27/0023/V.4

NUREG/CP-0023
Vol. 4

Proceedings of the U.S. Nuclear Regulatory Commission

Eighth Water Reactor Safety Research Information Meeting

NEWARK PUBLIC LIBRARY

APR 1 1982

U. S. GOVT. PUBS.
DEPOSITORY 376

Held at
National Bureau of Standards
Gaithersburg, Maryland
October 27-31, 1980

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

READEX FICHE best copy available

V 3 N88

The views expressed in these proceedings are not necessarily those of the U. S. Nuclear Regulatory Commission.

The submitted manuscript has been authored by a contractor of the U.S. Government under contract. Accordingly the U.S. Government retains a nonexclusive, royalty-free license to publish or reproduce the published form of this contribution, or allow others to do so, for U.S. Government purposes.

Available from

GPO Sales Program
Division of Technical Information and Document Control
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Printed copy price: $13.00

and

National Technical Information Service
Springfield, VA 22161

NUREG/CP-0023
Vol. 4

Proceedings of the U.S. Nuclear Regulatory Commission

Eighth Water Reactor Safety Research Information Meeting

Held at
National Bureau of Standards
Gaithersburg, Maryland
October 27-31, 1980

Date Published: March 1982

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

FINAL AGENDA

EIGHTH WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING

AT THE

NATIONAL BUREAU OF STANDARDS
ADMINISTRATION BUILDING 101
GAITHERSBURG, MARYLAND

October 27-31, 1980

TABLE OF CONTENTS

MONDAY, OCTOBER 27, 1980

9:15 am -	Introductory Remarks	Commissioner
9:30 am -	Reactor Safety Research Program	T. E. Murley, NRC
10:00 am -	Highlights of WRSR Achievement in FY 80 and Status of LOCA Safety Evaluation	L. S. Tong, NRC

LOFT PROGRAM

Chairman: G. D. McPherson, NRC

11:20 am -	Results of LOFT Small Break Experiments L3-1, L3-2, L3-5/5a, and L3-7.	J. H. Linebarger, INEL
12:10 pm -	LOFT: A Nuclear Plant Providing Realistic Answers to PWR Licensing Issues	C. W. Solbrig, INEL
12:30 pm -	Results of Anticipated Transient Experiments	C. W. Solbrig, INEL
2:00 pm -	Flow Measurement Techniques in LOFT Small Break Experiments	D. J. Hanson, INEL
2:35 pm -	LOFT Program Overview	N. C. Kaufman, INEL

SEMISCALE PROGRAM

Chairman: W. C. Lyons, NRC

3:30 pm -	Key Results of Semiscale	L. P. Leach, INEL

MONDAY, OCTOBER 27, 1980

SEMISCALE PROGRAM Cont'd

4:15 pm	Results of Semiscale Pumps On/Off Experiments	G. W. Johnsen, INEL

FUEL BEHAVIOR RESEARCH

Chairman: M. L. Picklesimer, NRC

9:15 am - 10:20 am	Introduction Research Highlights	See Page 1 of the Agenda
11:00 am	FRAPCON-2 Steady State Fuel Code	C. Mohr, PNL
11:30 am	Recent PNL Studies on GAP Conductance and Fuel Stored Energy	D. D. Lanning, PNL
12:00 pm	Status of Multi-Rod Burst Test Program	R. Chapman, ORNL
12:30 pm	Clad Embrittlement Criteria-Application To An Assessment of of the Margin of Performance of ECCS in LWRs	T. F. Kassner H. M. Chung, ANL

Severe Fuel Damage and Core Melt Research

Chairman: R. R. Sherry, NRC

2:00 pm	An Assessment of the Influence of Surface Thermocouples on the Behavior of Nuclear Fuel Rods During a Large Break LOCA	T. R. Yackle, INEL
2:25 pm	Response of Preirradiated Fuel Rod Bundle During Reactivity Initiated Accident Test 1-4	P. E. MacDonald, INEL
3:55 pm	Flooding Experiments in Blocked Arrays (FEBA): Recent Results and Future Plans	P. Ihle, KfK, FRG
4:40 pm	FG Release from Irradiated Fuel	S. Gehl, ANL
5:00 pm	Measured Release of Radioactive Xenon, Krypton, and Iodine from UO_2 During Nuclear Operation and a Comparison with Release Models	A. D. Appelhans, INEL

TUESDAY, OCTOBER 28, 1980

ANALYSIS DEVELOPMENT PROGRAM

Chairman: S. Fabic, NRC

(A) Development of Detailed Advanced Codes for Systems Analysis

9:15 am - TRAC-PD2 Description and Application	R. Pryor, LASL
10:30 am - TRAC-BD1, Transient Reactor Analysis Code for Boiling Water Systems	F. Aguilar, INEL

(B) Development of Fast Running, Simplified Geometry Advanced Codes for Systems Analyses

11:00 am - TRAC-PFO and PF1	D. Liles, LASL
11:30 am - RELAP-5 - An Advanced Fast Running LWR Transient Analysis Code	

(C) Development and Application of Component Codes

12:30 pm - Some Recent Applications of the K-FIX Code	B. J. Daly, LASL

(D) System Code Assessment and Application

2:00 pm - Code Assessment for Nuclear Reactor Accident Analysis Programs	S. Fabic, NRC
2:20 pm - Code Assessment Results at LASL	T. Knight, LASL
2:50 pm - Assessment of TRAC-PIA Using Calculations for Integral Facilities	A. C. Peterson, INEL
3:30 pm - Code (JRAC-PIA) Assessment Results at BNL	P. Saha, BNL

SEPARATE EFFECTS PROGRAM

Thermalhydraulic Modeling

Chairman: L. H. Sullivan, NRC

4:00 pm - Overview of Thermalhydraulic Modeling	L. H. Sullivan, NRC
4:05 pm - Steam-Generator Flow Patterns and Modeling	P. Griffith, MIT
4:25 pm - Upper Plenum Dump	Y. Sudo, JAERI

TUESDAY, October 28, 1980
SEPARATE EFFECTS PROGRAM
Thermalhydraulic Modeling Cont'd

4:40 pm - Summary of Condensation Studies	S. G. Bankoff, NWU
5:05 pm - RPI Research in the Area of Phase Distribution and Separation Phenomena and LWR Instability Phenomena	R. T. Lahey, Jr. RPI

RISK AND SEVERE ACCIDENT SEQUENCE ANALYSIS

Chairman: Gordon E. Edison, NRC

9:45 am - Auxiliary Feedwater Reliability in PWRs	G. E. Edison, NRC
10:30 am - Analysis of Pressurized Water Reactor Station Blackout	B. F. Saffell, INEL
11:00 am - Loss-of-Feedwater Transients in PWRs	R. D. Burns, III, LASL
11:30 am - Severe Core Damage Accident Progression: Best Estimate and Uncertainties	W. B. Murfin J. B. Rivard M. L. Corradini, Sandia
12:00 pm - Severe Accident Sequence Assessment in BWRs	M. H. Fontana, ORNL

ADVANCED INSTRUMENTATION

Chairmen: Y. Y. Hsu and N. N. Kondic, NRC

2:00 pm - Overview	Y. Y. Hsu, NRC
2:10 pm - Pulsed Neutron Activation Techniques in Water Reactor Safety Research	P. Kehler, ANL
2:25 pm - A Pulsed Neutron Generator for Use With Pulsed Neutron Activation Techniques	G. E. Rochau, Sandia
2:40 pm - Liquid Level Detection Techniques	K. G. Turnage G. N. Miller, ORNL
3:25 pm - 2D/3D Instrumentation Overview - ORNL	B. Eads, ORNL
3:55 pm - Overview of 2D/3D Instrumentation Developed at EG&G Idaho, Inc.	R. E. Rice, INEL

TUESDAY, OCTOBER 28, 1980
ADVANCED INSTRUMENTATION cont'd

4:05 pm -	Low Energy Sodium Iodide Gamma Densitometer for 2D/3D Program	J. B. Colson, INEL
4:20 pm -	An Optical Liquid Level Detector for High Temperature/Pressure Water Environment	B. L. Watson, INEL
4:35 pm -	Advanced Instrumentation Project	W. H. Roach, INEL
4:45 pm -	Laser Doppler Anemometry Instrumentation of Two-Phase Flows	M. L. Wilson, INEL
5:00 pm -	Steam Generator Instrumentation	J. R. Wolf, INEL
5:20 pm -	Non-intrusive Density Profile Determination Gamma Beam Densitometer, Tomography and Scattering	N. N. Kondic, NRC

WEDNESDAY, OCTOBER 29, 1980

2D/3D RESEARCH PROGRAM

Chairman, W. S. Farmer, NRC

9:30 am -	Results of PKL Small Break Experiments	D. Hein F. Winkler, KWU, FRG
10:25 am -	The German 2D/3D-UPTF Program	E. F. Hicken K. Hofmann, GRS, FRG
10:50 am -	Results of CCTF Core 1 Tests	Y. Murao, JAERI
11:30 am -	TRAC Analysis Support for the 2D/3D Program	K. A. Williams, LASL
12:10 pm -	Measurement of Two-phase Flow at the Core Upper Plenum Interface Under Simulated Reflood Conditions	D. G. Thomas, ORNL

SEPARATE EFFECTS PROGRAM

Chairman: W. D. Beckner, NRC

2:00 pm -	THTF Heat Transfer Data	J. D. White, ORNL

WEDNESDAY, OCTOBER 29, 1980
SEPARATE EFFECTS PROGRAM Cont'd

2:45 pm -	BWR Blowdown/Emergency Core Cooling Integral Program (TLTA Large and Small Break)	G. L. Sozzi, GE
3:30 pm -	BWR Refill-Reflood Program: Overview and Experimental Results	G. W. Burnette, GE
4:00 pm -	BWR Refill-Reflood Program Model Development for TRAC-BD	J. G. M. Andersen, GE
4:30 pm -	FLECHT-SEASET (1) Unblocked Channel Data (2) Blocked 21-Rod Bundle Tests	L. E. Hochreiter, W

REACTOR OPERATIONAL SAFETY PROGRAM

Chairman: R. Feit, NRC

10:45 am -	Status of the Fire Protection Program	L. J. Klamerus, Sandia
11:30 am -	Fire Protection System Modeling: The Fire Resistance of Walls Penetrated by Electric Cables	L. W. Hunter S. Favin, APL

REACTOR OPERATIONAL SAFETY: OPERATOR-MACHINE INTERFACE

Chairman: W. S. Farmer, NRC

2:00 pm -	Advances in Noise Analysis for Nuclear Plant Surveillance and Diagnostics	D. N. Fry, ORNL
2:30 pm -	The Safety-Related Operator Actions Program at ORNL	P. M. Haas, ORNL
3:30 pm -	Simulators and Thier Use in Training Operators	D. W. Jones, MSU/CNS
4:00 pm -	Defining the Role of the Operating Crew	R. A. Kisner, ORNL
4:30 pm -	Advanced Display and Diagnostic at LOFT	O. R. Meyer, INEL

WEDNESDAY, OCTOBER 29, 1980

EPRI REACTOR SAFETY RESEARCH PROGRAM

2:00 pm - Safety Research at EPRI-An Update	W.B. Loewenstein, EPRI
3:30 pm - Recent Missile Tests	G. Sliter, EPRI
4:00 pm - BWR IGSCC Research Program	K. Stahlkopf, EPRI
4:30 pm - Analysis of Small-Break Heat Removal Tests	R. Duffey, EPRI
5:00 pm - Validating Risk Analysis; Selected Aspects	G. Lellouche, EPRI

THURSDAY, OCTOBER 30, 1980

APPLIED MECHANICS AND SITE TECHNOLOGY

Seismic Safety Margins Research Program

9:50 am - Introduction	J. Richardson, NRC
10:00 am - Overview of SSMRP Program	P. Smith, LLL
10:40 am - Systems Model and Methodology	G. Cummings, LLL
11:20 am - Seismic Input	D. Bernreuter, LLL
12:00 pm - Soil-Structure Interaction	J. Johnson, LLL

Seismic Safety Margins Research Program

Chairman: C. W. Burger, NRC

2:00 pm - Structural Response	J. Johnson, LLL
2:40 pm - Subsystem Response	T. Y. Chuong, LLL
3:30 pm - Component Fragility	M. Bohn, LLL
4:30 pm - Best Estimate vs Evaluation Model	J. Johnson, LLL

THURSDAY, OCTOBER 30, 1980
SITE SAFETY RESEARCH PROGRAM
Meteorology

Chairman: R. F. Abbey, NRC

9:50 am	Overview of NRC Meteorology Research Program	R. F. Abbey, NRC
10:45 am	Near-Ground Tornado Wind Fields	J. R. McDonald, Texas Tech. U
11:15 am	Measured Pressure Loads on Model Structures in Simulated Tornado-Like Flow	M.C. Jischke, U of Oklahoma
11:45 am	Automobile Impact Studies	R. Chiapetta, Chiapetta-Welch & Assoc.
12:15 pm	Atmospheric Dispersion Field Experiments to 80 km	I. Van der Hoven, NOAA Air Resources Laboratory

Seismic Hazard

Chairman: J. Harbour, NRC

2:00 pm	Overview of NRC Programs in Seismology and Geology	J. Harbour, NRC
3:00 pm	Reservoir-Induced Seismicity	A. Murphy, NRC
3:45 pm	Northeastern U.S. Seismic Network	P. Pomeroy, NRC Consultant
4:30 pm	Tectonic Features in the Vicinity of the Charleston 1886 Earthquake	J. Behrendt, USGS

METALLURGY AND MATERIALS RESEARCH PROGRAMS

9:15 am	Welcome and Introduction	L. C. Shao, NRC
10:00 am	Introduction	C. Z. Serpan, NRC

Irradiation Effects and Neutron Dosimetry

Chairman: C. Z. Serpan, NRC

10:30 am	LWR Dosimetry Improvement Program Overview	W. N. McElroy, HEDL
11:00 am	Reactor Calculation "Benchmark" PCA Blind Tests Results	F.B.K. Kam, ORNL

THURSDAY, OCTOBER 30, 1980
METALLURGY AND MATERIALS RESEARCH PROGRAMS
Irradiation Effects and Neutron Dosimetry Cont'd

11:30 pm -	Neutron Characterization of HSST Irradiation Facility and of Simulated RPV Dosimetry - Embrittlement Experiment	A. Fabry, ORNL/CEN/SCK

Fracture Mechanics

Chairman: M. Vagins, NRC

12:00 pm -	Experimental Verification of the Behavior of Surface Flaws in Thick-Walled Steel Cylinders During Severe Thermal Shock (TSE-5 and TSE-5A)	R. Cheverton, ORNL
12:30 pm -	Crack Stability Analysis for Vessel Tests in the Upper Shelf Temperature Range	J. Merkle, ORNL
2:00 pm -	Validation of "Key Curve" Analysis of Elastic-Plastic Fracture Toughness	J. Joyce, USNA
2:30 pm -	Toughness and Ductile Shelf Properties of Irradiated Low-Shelf Weld Metals	F. J. Loss, NRL
3:20 pm -	Cyclic Irradiation - Annealing - Reirradiation of RPV Steels and Welds	J. R. Hawthorne, NRL
3:50 pm -	Crack Growth Rate of Irradiated Vessel and of Piping Steels in PWR Environments	H. Watson/ W. Cullen, NRL
4:20 pm -	Crack Arrest Methodology and and Standard Test Methods for RPV Evaluations	G. Irwin/W. Fourney, UMD
4:50 pm -	Validation of Tearing Instability on Degraded LWR Piping	J. P. Gudas, NSRDC

FRIDAY, OCTOBER 31, 1980
Pipe Failure Model and Effects

Chairman: M. Vagins, NRC

9:15 am -	Probability Models for Piping Failure	D. Harris, SAI

FRIDAY, OCTOBER 31, 1980
METALLURGY AND MATERIALS RESEARCH PROGRAMS
Pipe Failure Models and Effects Cont'd

9:55 am -	Experimental Program for Pipe to Pipe Impact Effects	M. C. C. Bampton, PNL
10:30 am -	Verification of Two Phase Jet	D. Tomasko, Sandia
11:00 am -	Pipe Whip Code Development	G. Powell, Un. of CA/ Berkeley

Environmentally Assisted Cracking and Steam Generator Integrity

Chairman: J. Muscara, NRC

11:30 am -	Program for Environmental-Assisted Cracking in LWRs	W. J. Shack, ANL
12:10 am -	Progress and Plans for Steam Generator Integrity Research	R. Clark, PNL
12:40 am -	Stress-Corrosion Cracking of Steam Generator Tubes	D. VanRooyen, BNL

Nondestructive Evaluation

Chairman: J. Muscara, NRC

2:00 pm -	Improved Eddy Current Inspection of Steam Generator Tubes	C. V. Dodd, ORNL
2:25 pm -	Reliability of Flaw Detection	F. L. Becker, PNL
3:20 pm -	SAFT-UT for Flaw Imaging and Display Techniques	G. Ganapathy, U Mich
3:55 pm -	ISI Application of SAFT-UT	J. Jackson, SWRI
4:25 pm -	Models for A/E Monitoring of Reactors	P. H. Hutton, PNL
4:55 pm -	Detection of IGSCC Initiation	L. Yeager, DAI

STRUCTURAL ENGINEERING RESEARCH PROGRAMS

Chairman: G. Bagchi, NRC

9:15 am -	Introduction	G. Bagchi, NRC
9:30 am -	Category I Structures (Safety Margin at Ultimate Load)	C. A. Anderson, LASL

FRIDAY, OCTOBER 31, 1980

STRUCTURAL ENGINEERING RESEARCH PROGRAMS Cont'd

10:30 am - Hydrogen Explosion	M. Fardis, MIT
11:15 am - Containment Safety Margin	W. VonRiesemann, Sandia
12:00 pm - Evaluation of Dynamic Testing of NPP Structures	C. A. Kot, M. G. Srinivasan, ANL

STRUCTURAL ENGINEERING RESEARCH BRANCH

Chairman: G. Bagchi, NRC

2:00 pm - Codes and Standards With Relation to Containment Safety Margins	R. N. White, Cornell
2:40 pm - Large Scale Testing of Containment Elements	H. G. Russell, PCA
3:30 pm - Analytical Approach for Evaluation of Codes, Standards and the Inherent Safety Margin in Safety-Related Structures	J. J. Connor, MIT
4:15 pm - Load Combinations	B. Ellingswood, NBS

MECHANICAL ENGINEERING RESEARCH PROGRAM

Load Combinations

Chairman: J. O'Brien, NRC

9:15 am - Introduction	J. A. O'Brien, NRC
9:25 am - General Description of Load Combination Program	C. K. Chou, LLNL

Event Decoupling (LOCA Plus Earthquake)

9:45 am - Probabilistic Model and Computational Procedure	L. L. George, LLNL
10:20 am - Fracture Mechanics Evaluation	R. D. Streit, LLNL
10:50 am - Results and Conclusion	S. C. Lu, LLNL

FRIDAY, OCTOBER 31, 1980

MECHANICAL ENGINEERING RESEARCH PROGRAMS
Load Combinations Cont'd

Load Combination Methodology

11:25 am	- Load Combination Methodology	C. A. Cornell, MIT
11:55 am	- Load Combination Methodology	M. W. Schwartz, LLNL
12:15 pm	- Application to Nuclear Systems of the Load Combination Methodology	M. K. Ravindra, S&L

Chairman: D. Reiff, NRC

2:00 pm	- Nonlinear System Modeling	S. Masri, USC
2:30 pm	- Piping Benchmarks	M. Reich, BNL
3:15 pm	- Snubber Research and Testing	A. Onesto, ETEC
3:45 pm	- Experimental and Preliminary Analytical Results of Coupled Fluid-Structure Interactions During Blowdown of the HDR Vessel	U. Schumann, KfK (FRG)
4:10 pm	- Earthquake Simulation Experiments Performed at the HDR Facility	G. Katzemeier, KfK (FRG)
4:30 pm	- HDR Dynamic Tests of Late 1979	G. Howard, ANCO
5:00 pm	- Predictions of Recirculation Loop Response at HDR to Simulated Blast Excitation	R. Guenzler, INEL

PROBABILITY MODELS FOR PIPING FAILURE

D.O. HARRIS
SCIENCE APPLICATIONS, INC.

The purpose of this project is to develop a model for prediction of piping reliability in light water reactors, and to apply the model to issues of concern to the NRC in regards to the safety of nuclear plants. Specifically, the model has been applied to the question of the probability of a seismic reduced LOCA in the primary loop of a pressurized water reactor. Results obtained should be of use in making decisions whether to require nuclear plants to be designed for a simultaneous seismic event and complete pipe severance in the primary loop. This work was performed as part of the Lawrence Livermore Laboratory Load Combinations Program, which in turn was funded by the NRC.

The probability model for piping failure will also be applied to the question of the failure probability as a function of position within various piping systems in LWR's. Such results will provide a rational basis for postulating pipe break locations. This will serve as a rational means of revising Reg. Guide 1.46 to account for the most likely break locations.

The probability model for piping failure is based on fracture mechanics calculations of the growth of cracks that are considered to be statistically distributed within the piping weldments. Currently, part-circumferential interior surface cracks are being considered. Material fracture mechanics characteristics, failure criteria, and inspection detection probability are included in the model, along with plant operating history of cyclic stresses and numbers of cycles. A bivariate crack size distribution is considered, and pipe failure due to leaks and complete pipe severances are distinguished.

The model has been applied to the analysis of the probability of failure in the circumferential welds of the main coolant piping of a PWR. As expected, the calculated leak failure probabilities are several orders of magnitude higher than the probability of a complete pipe severance. However, the complete pipe severance probabilities are quite low, which is most likely because only failures due to the fatigue growth of pre-existing cracks under a stress history that includes only anticipated transients and seismic events are considered. The results to date indicate that a pre-service inspection and proof test are influential in reducing the calculated failure probability. In general, seismic events and in-service inspection of the austenitic weldments have only a minimal influence on the calculated failure probabilities. Results anticipated within the next year will generalize these results to other piping systems and materials, and provide guidance in possible revisions of regulatory guides.

PROBABILITY MODELS OF PIPING FAILURE

- SAFETY ISSUES ADDRESSED

  — WHAT IS THE LIKELIHOOD OF A SEISMIC-INDUCED COMPLETE PIPE SEVERANCE, AND SHOULD REACTOR COMPONENTS BE REQUIRED TO BE DESIGNED FOR A SIMULTANEOUS OCCURRANCE OF SUCH LOADS?

  — WHAT ARE THE MOST LIKELY LOCATIONS FOR BREAKS IN VARIOUS REACTOR PIPING SYSTEMS?

- RESULTS OBTAINED OR EXPECTED

  — OF THE COMPLETE PIPE SEVERANCES IN THE MAIN COOLANT PIPING OF A PWR, ONLY 4 IN 100 WILL BE IN COMBINATION WITH A SEISMIC EVENT (I.E. SEISMIC-INDUCED)

  — CURRENT MODEL WILL BE EXPANDED AND EXERCIZED TO CALCULATE RELATIVE PROBABILITY OF FAILURE AT VARIOUS LOCATIONS IN VARIOUS REACTOR PIPING SYSTEMS.

MODEL DESCRIPTION

ASSUMPTIONS: IN ASSESSING HOW REALISTIC THE MODEL RESULTS ARE, IT IS IMPORTANT TO REMEMBER THE ASSUMPTIONS.

- FAILURE OCCURS ONLY AS A RESULT OF THE GROWTH OF CRACK-LIKE DEFECTS INTORDUCED DURING FABRICATION.

- THE AS-FABRICATED DEFECT GROWS AS A FATIGUE CRACK DUE TO CYCLIC LOADS IMPOSED DURING PLANT OPERATION.

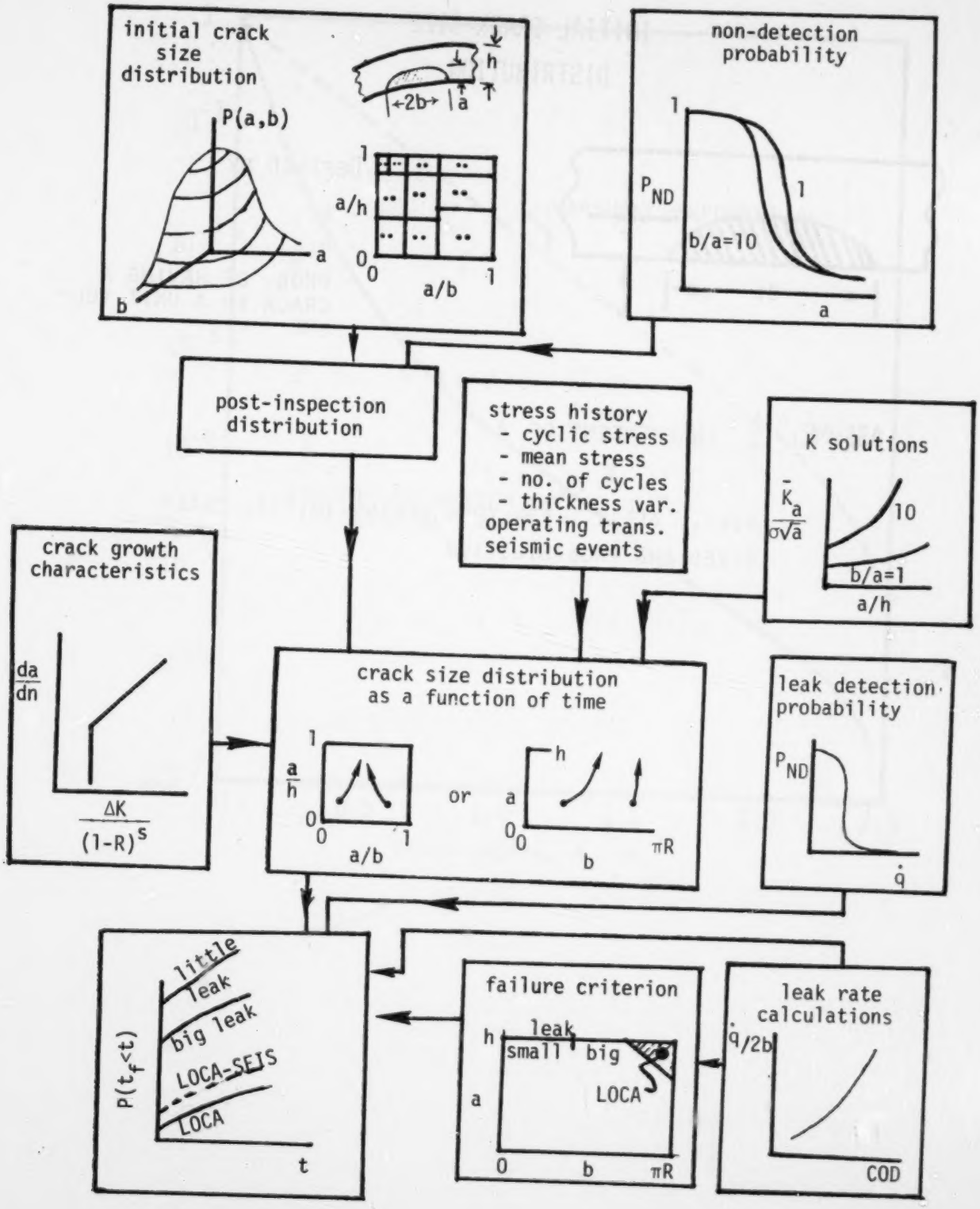
- THE DEFECT GROWTH RATES CAN BE PREDICTED FROM LABORATORY FRACTURE MECHANICS TESTS (I.E. DA/DN VS. ΔK)

- THE IMPOSED CYCLIC STRESS HISTORY IS COMPOSED OF SEISMIC EVENTS AND ANTICIPATED TRANSIENTS OF ANTICIPATED CYCLIC STRESS MAGNITUDE.

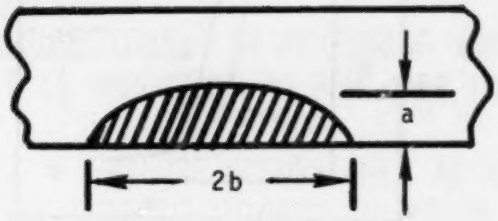
- THE PIPING MATERIALS AND WELD PROCEDURES ARE ACTUALLY AS SPECIFIED.

- WATER CHEMISTRY AND PLANT OPERATION ARE AS ANTICIPATED.

- NO DESIGN ERRORS HAVE OCCURRED.


initial crack size distribution
P(a,b)
a
b
h
2b
a
1
a/h
0
0
a/b
1
non-detection probability
1
P ND
1
b/a=10
a
post-inspection distribution
stress history
- cyclic stress
- mean stress
- no. of cycles
- thickness var.
operating trans.
seismic events
K solutions
K a / σ√a
10
b/a=1
a/h
crack growth characteristics
da/dn
ΔK / (1-R)^s
crack size distribution as a function of time
1
a/h
0
0
a/b
1
or
h
a
0
0
b
πR
leak detection probability
P ND
q̇
little leak
big leak
LOCA-SEIS
LOCA
P(t_f<t)
t
failure criterion
h
leak
small
big
LOCA
a
0
b
πR
leak rate calculations
q̇/2b
COD

INITIAL CRACK SIZE DISTRIBUTION

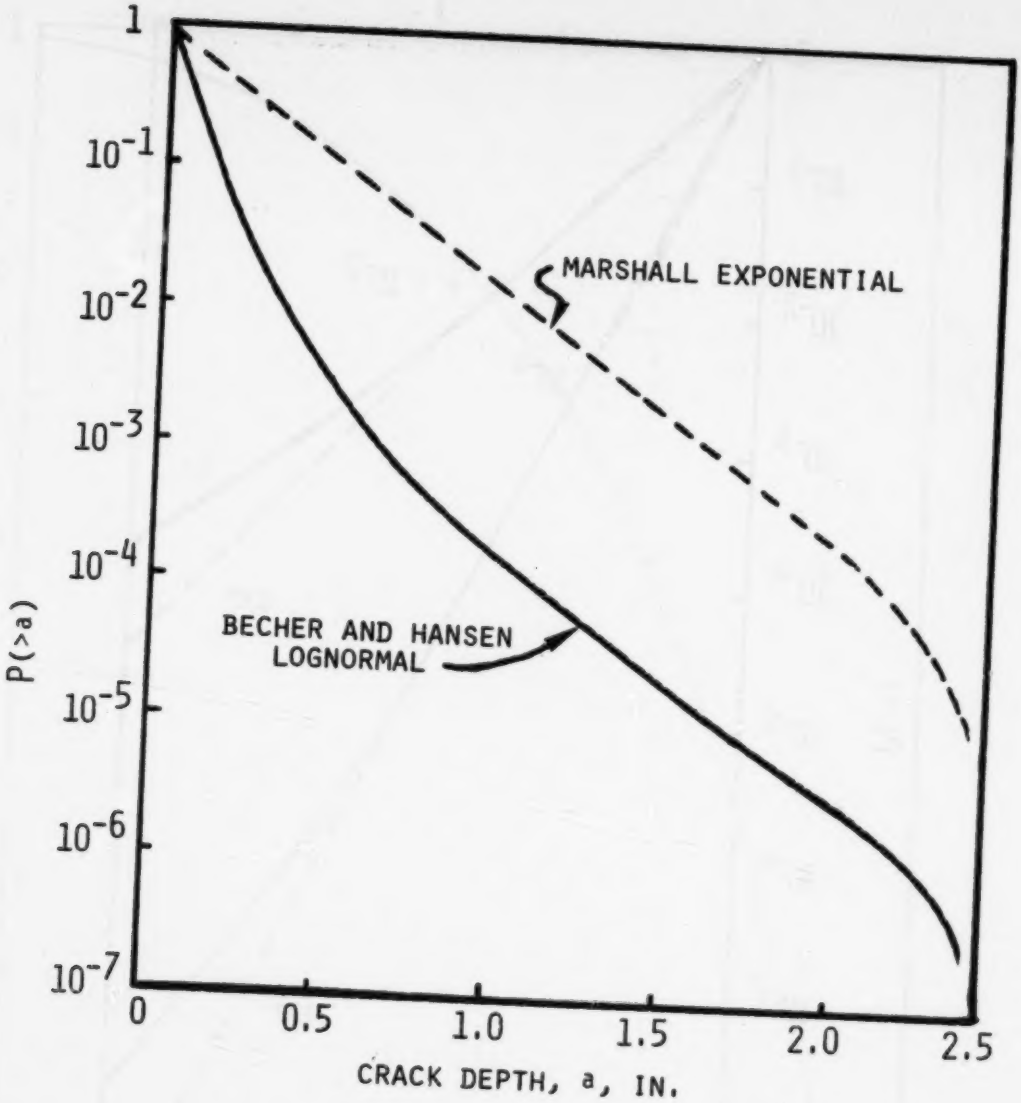


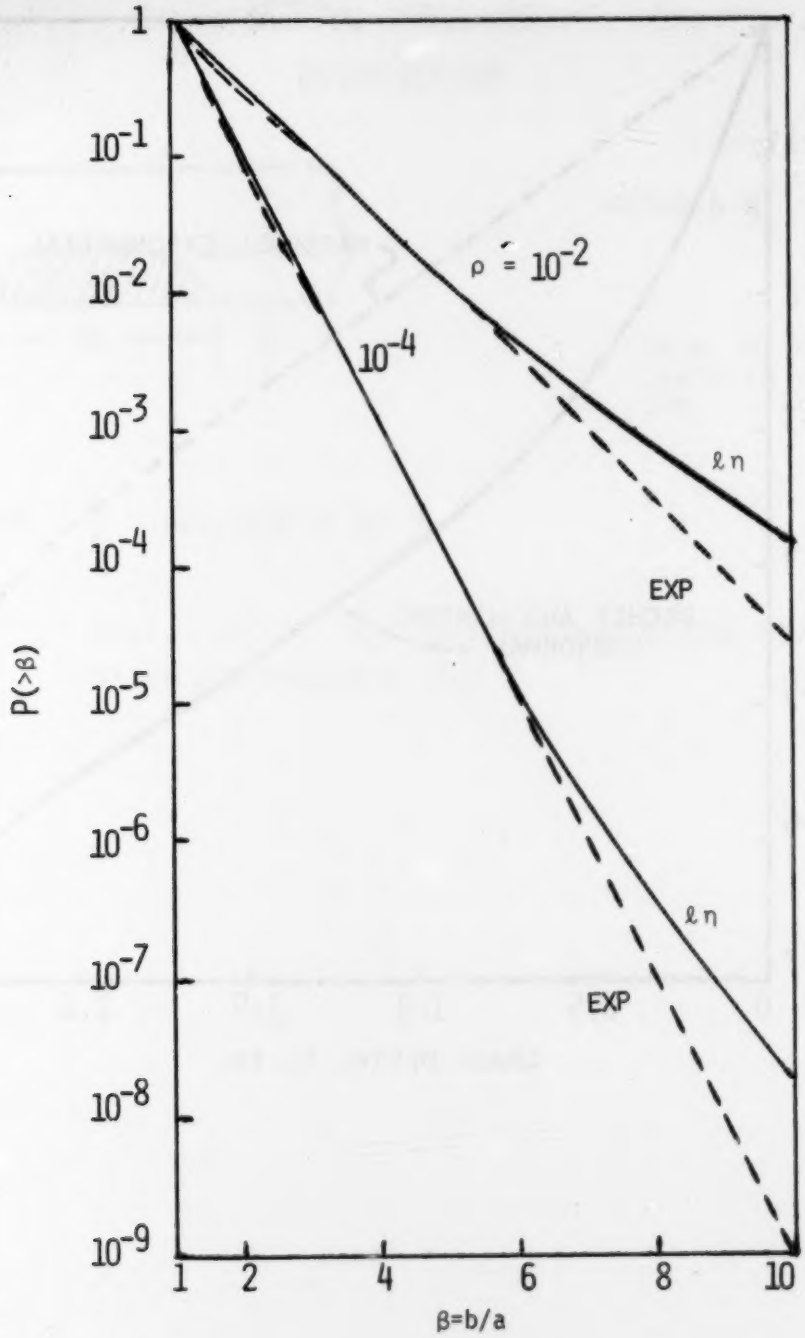
DEFINED BY

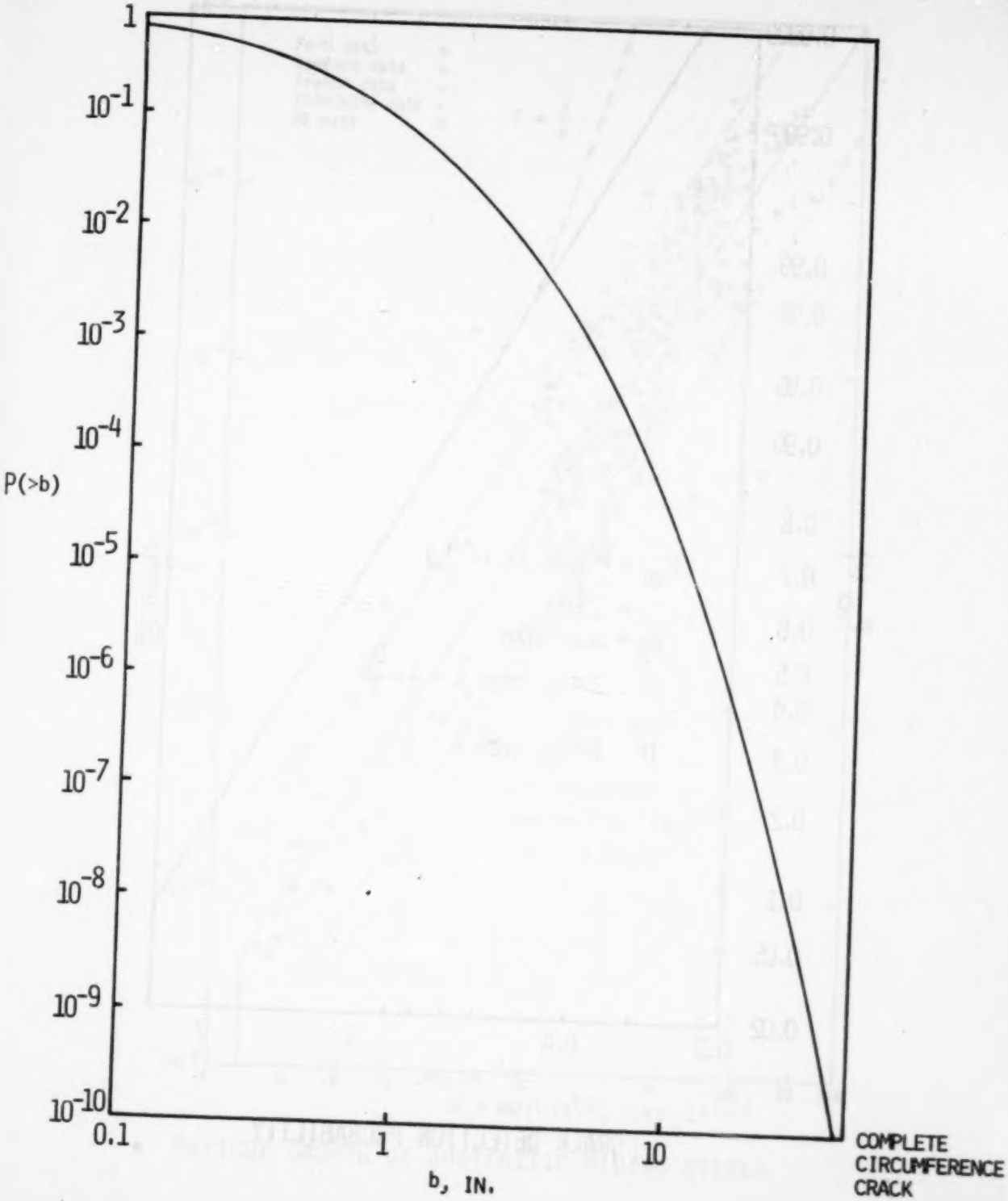
- a
- b/a
- PROB. OF HAVING A CRACK IN A UNIT VOLUME

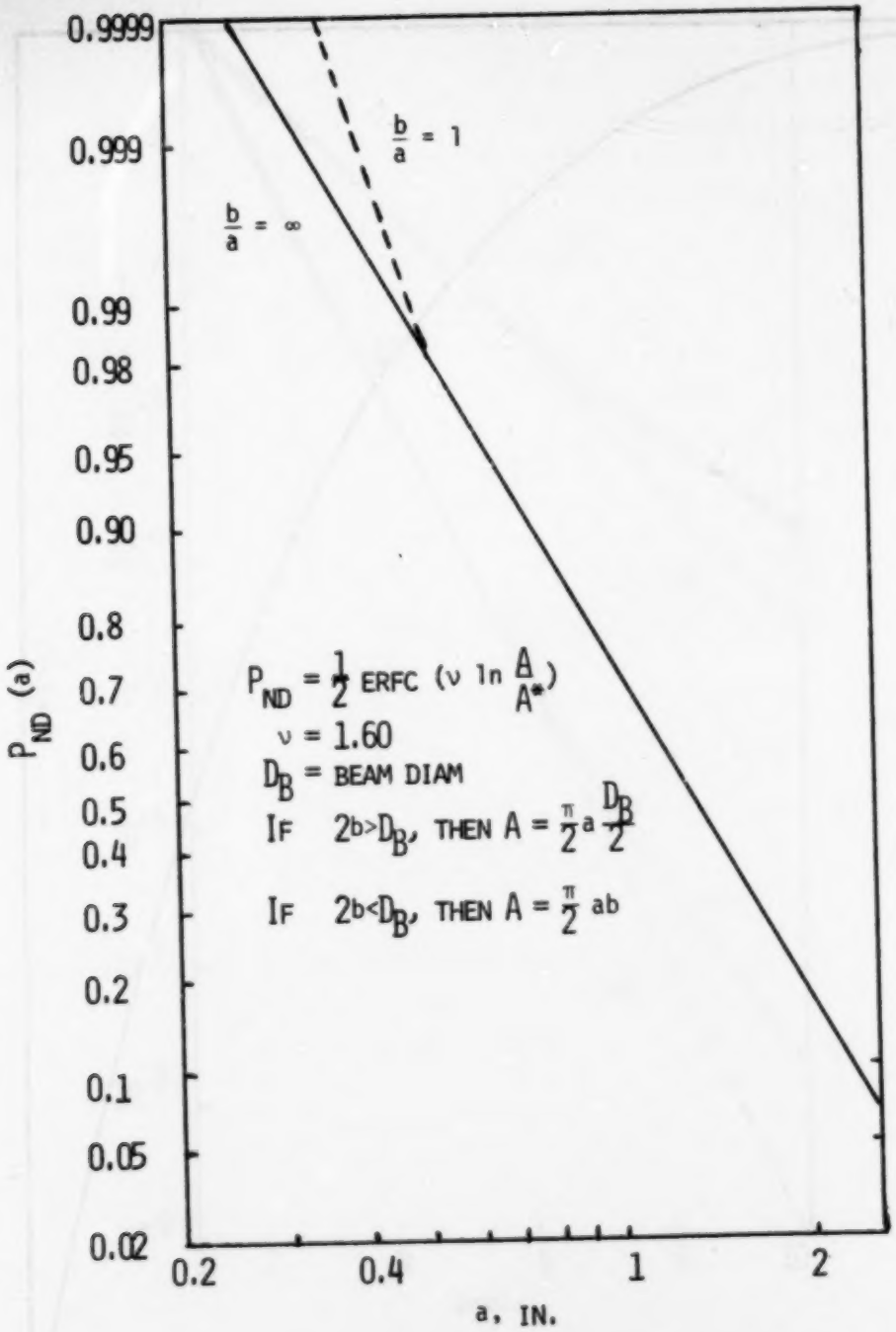
ASSUME: $\frac{b}{a}$ INDEPENDENT OF a.

$p(a)$, $p(\frac{b}{a})$, $p_v{}^*$ THEN DEFINE INITIAL CRACK SIZES AND PROBABILITIES

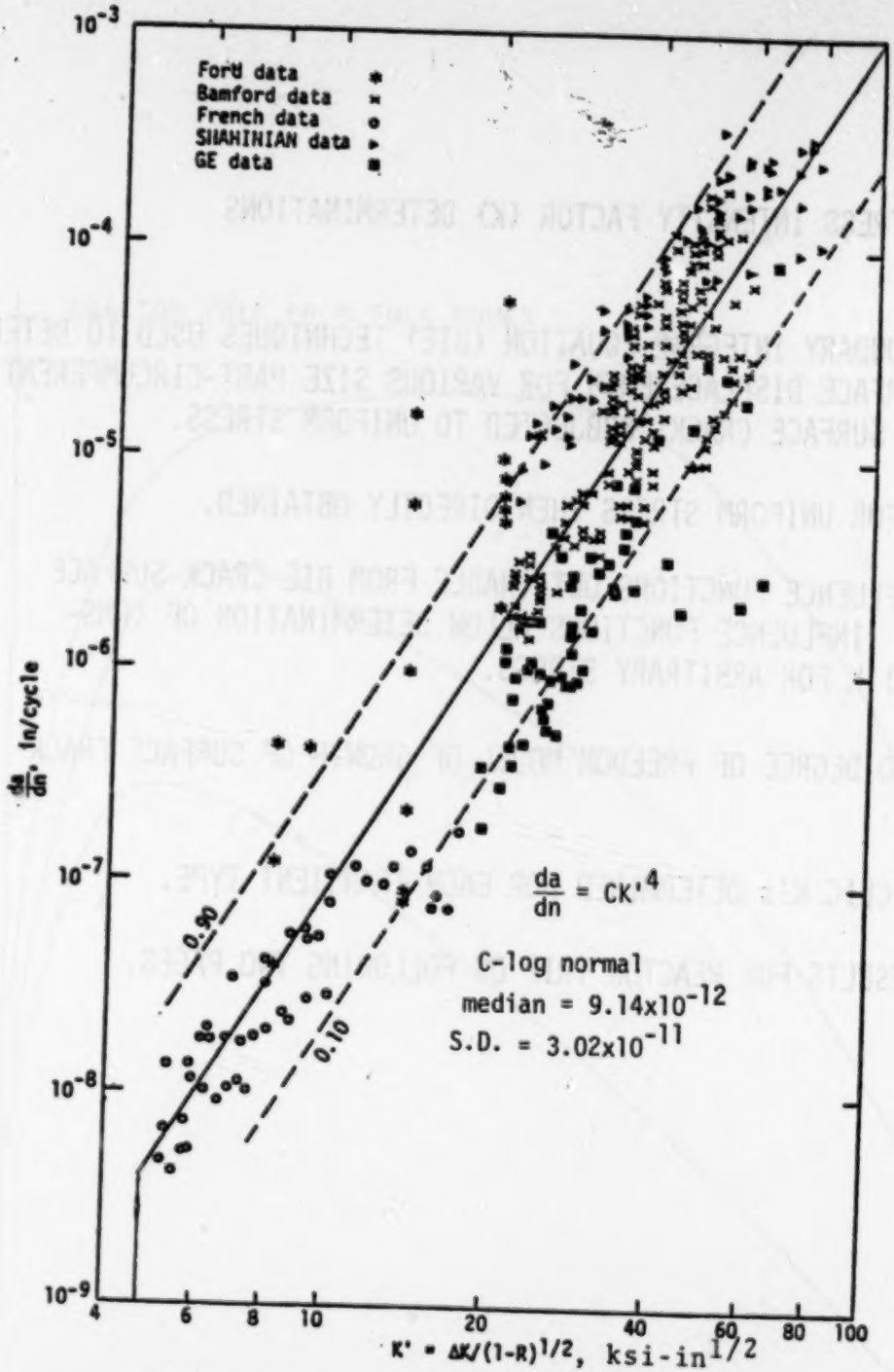

1
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-6}
10^{-7}
P(>a)
MARSHALL EXPONENTIAL
BECHER AND HANSEN
LOGNORMAL
0
0.5
1.0
1.5
2.0
2.5
CRACK DEPTH, a, IN.


1
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-6}
10^{-7}
10^{-8}
10^{-9}
P(>β)
$\rho = 10^{-2}$
10^{-4}
ℓn
EXP
ℓn
EXP
1
2
4
6
8
10
β=b/a


1
10^{-1}
10^{-2}
10^{-3}
10^{-4}
P(>b)
10^{-5}
10^{-6}
10^{-7}
10^{-8}
10^{-9}
10^{-10}
0.1
1
10
b, IN.
COMPLETE
CIRCUMFERENCE
CRACK



CRACK DETECTION PROBABILITY



- FATIGUE GROWTH OF AUSTENITIC PIPING STEELS

STRESS INTENSITY FACTOR (K) DETERMINATIONS

- BOUNDARY INTEGRAL EQUATION (BIE) TECHNIQUES USED TO DETERMINE CRACK SURFACE DISPLACEMENTS FOR VARIOUS SIZE PART-CIRCUMFERENTIAL INTERIOR SURFACE CRACKS SUBJECTED TO UNIFORM STRESS.

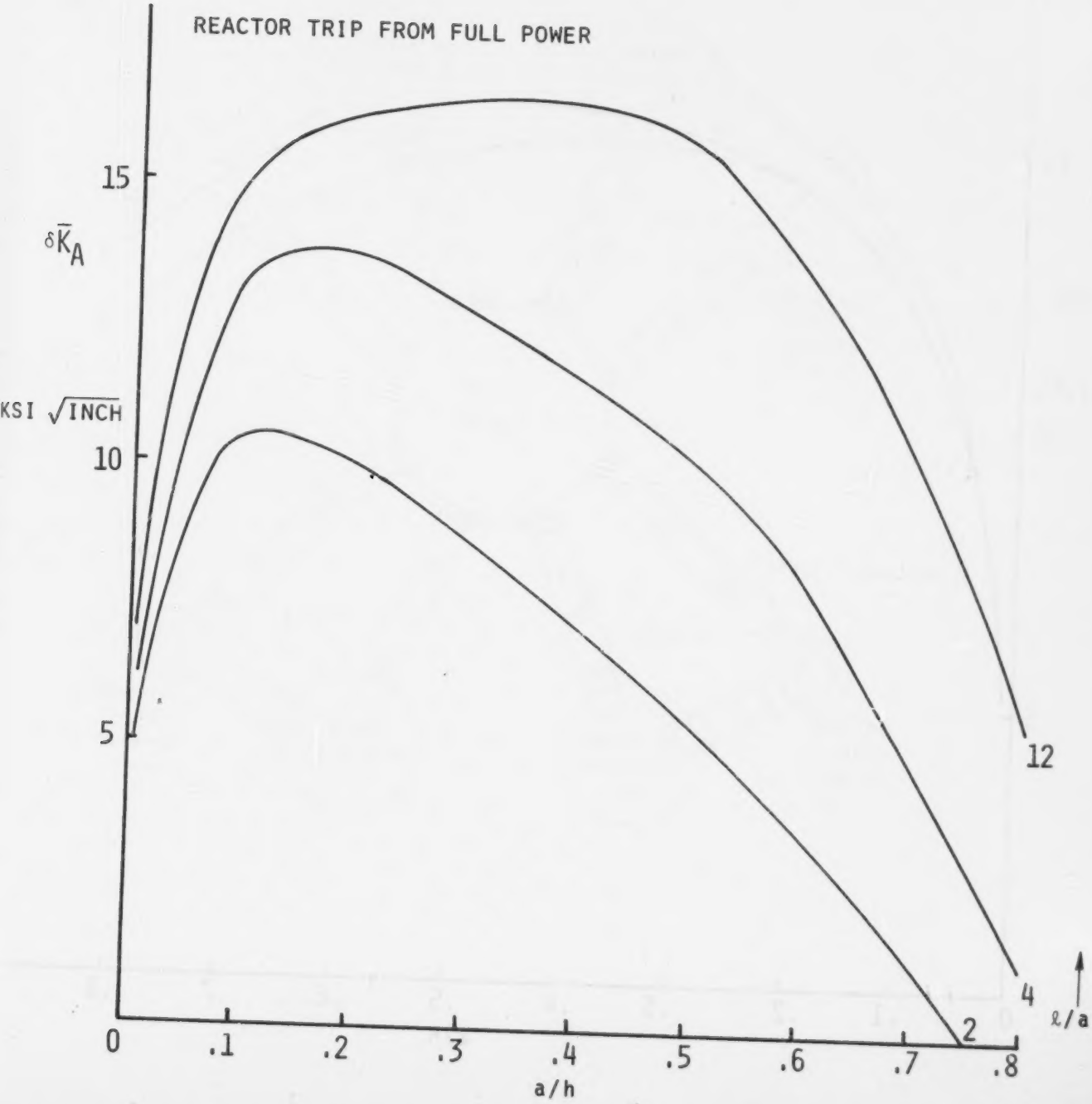
- K FOR UNIFORM STRESS THEN DIRECTLY OBTAINED.

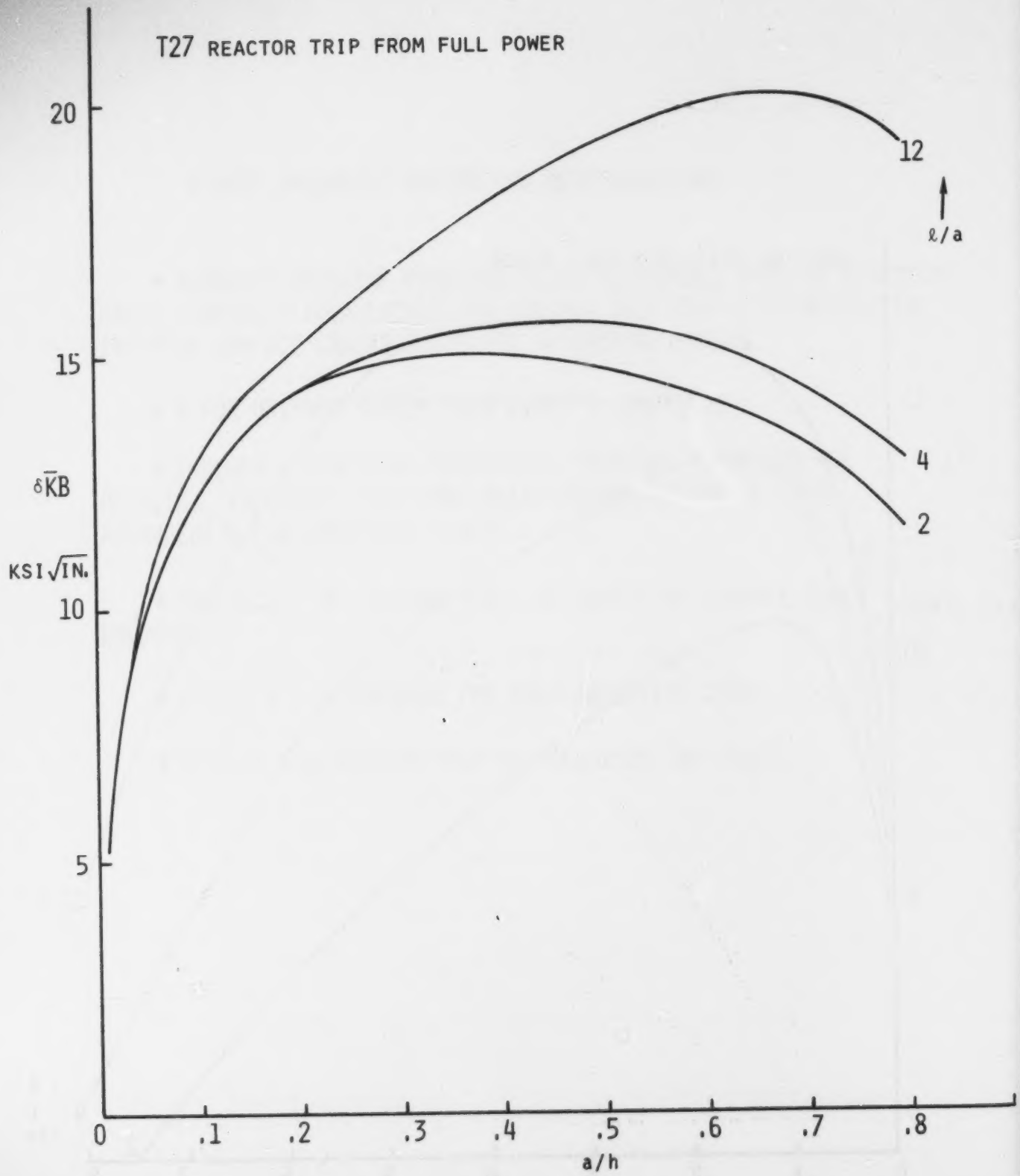
- INFLUENCE FUNCTIONS OBTAINABLE FROM BIE CRACK SURFACE RESULTS. INFLUENCE FUNCTIONS ALLOW DETERMINATION OF (RMS-AVERAGED) K FOR ARBITRARY STRESS.

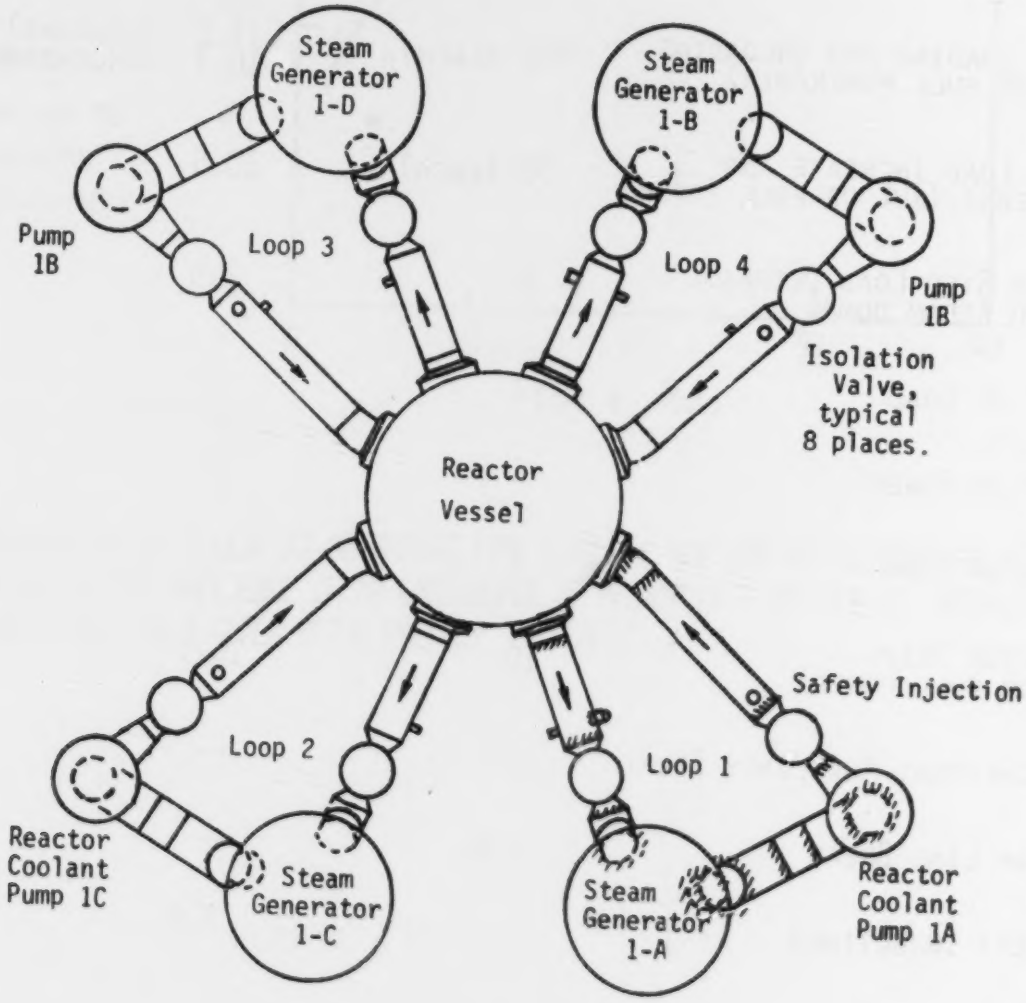
- TWO DEGREE OF FREEDOM MODEL OF GROWTH OF SURFACE CRACK EMLPOYED.

- CYCLIC K's DETERMINED FOR EACH TRANSIENT TYPE.

- RESULTS FOR REACTOR TRIP ON FOLLOWING TWO PAGES.


REACTOR TRIP FROM FULL POWER
15
$\delta\bar{K}_A$
KSI $\sqrt{\text{INCH}}$
10
5
12
4
2
ℓ/a
0
.1
.2
.3
.4
.5
.6
.7
.8
a/h

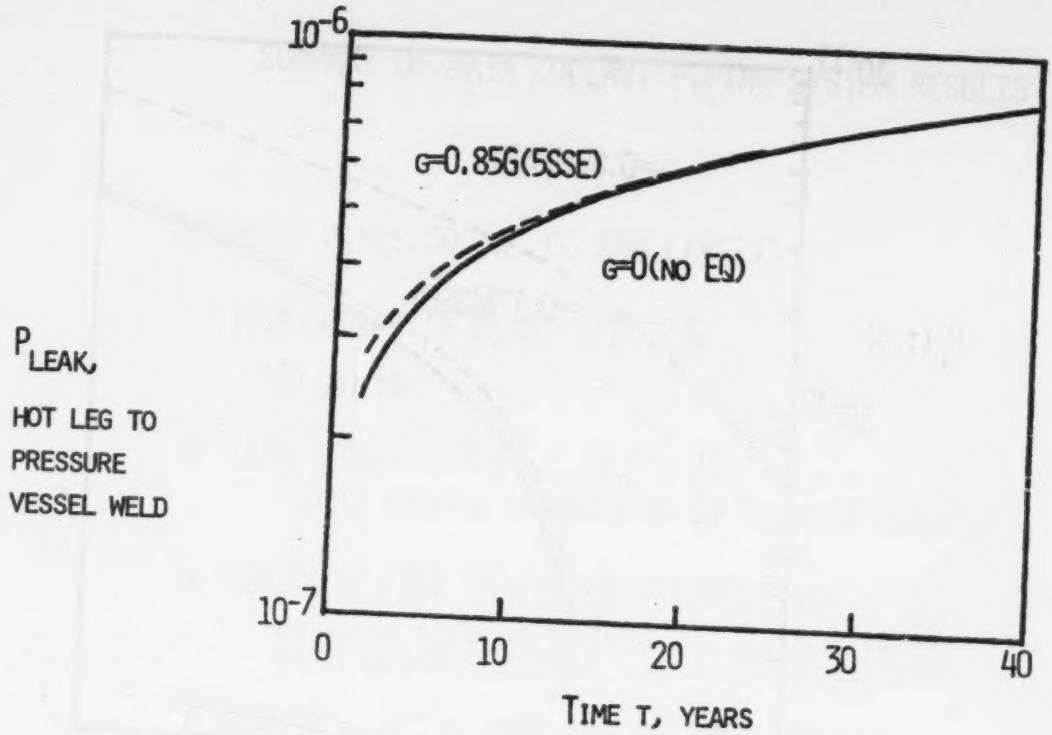

T27 REACTOR TRIP FROM FULL POWER
20
15
10
5
δK̄B
KSI √IN.
12
4
2
ℓ/a
0
.1
.2
.3
.4
.5
.6
.7
.8
a/h


Steam Generator 1-D
Steam Generator 1-B
Pump 1B
Loop 3
Loop 4
Pump 1B
Isolation Valve, typical 8 places.
Reactor Vessel
Safety Injection
Loop 2
Loop 1
Reactor Coolant Pump 1C
Steam Generator 1-C
Steam Generator 1-A
Reactor Coolant Pump 1A

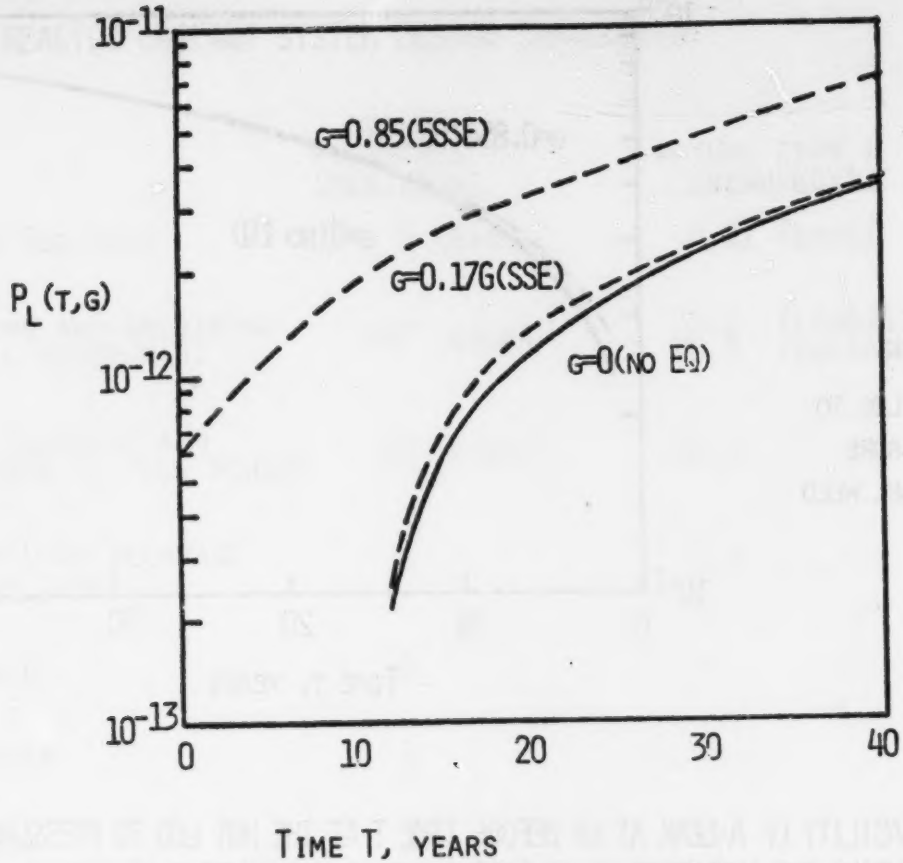
REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

	DESIGN FREQUENCY (PER YEAR)	ACTUAL ZION I FREQUENCY*
HEATUP AND COOLDOWN	5 (EACH)	4.91 (EACH)
UNIT LOADING AND UNLOADING (5% OF FULL POWER/MIN)	457 (EACH)	13.2 (LOADING) 10.3 (UNLOADING)
STEP LOAD INCREASE AND DECREASE (10% OF FULL POWER)	50 (EACH)	10.5
LARGE STEP LOAD DECREASE (WITH STEAM DUMP)	5	1.4
LOSS OF LOAD	2	--
LOSS OF POWER	1	--
LOSS OF FLOW	2	--
REACTOR TRIP (FROM FULL POWER)	10	16.3
INADVERTENT AUXILIARY SPRAY	.25	--
STEAM LINE BREAK	1/40	--
SAFETY INJECTIONS	--	3.4

* PER CALENDAR YEAR



PROBABILITY OF A LEAK AT OR BEFORE TIME T AT THE HOT LEG TO PRESSURE VESSEL WELD INCLUDING THE INFLUENCE OF AN EARTHQUAKE OF MAGNITUDE G AT TIME T. RESULTS ARE CONDITIONAL ON A CRACK BEING PRESENT INITIALLY.



PROBABILITY OF A COMPLETE PIPE SEVERANCE AT OR BEFORE TIME T AT THE HOT LEG TO PRESSURE VESSEL WELD INCLUDING THE INFLUENCE OF AN EARTHQUAKE OF MAGNITUDE G AT TIME T. RESULTS ARE CONDITIONAL ON A CRACK BEING PRESENT INITIALLY.

SUMMARY OF MAIN COOLANT PIPING SYSTEM RESULTS

(PRIMARY PIPES 30 IN. OD AND LARGER)

CUMULATIVE IN PLANT LIFETIME

- LEAK PROBABILITY : 9.4×10^{-7}
  (VERY WEAKLY INFLUENCED BY SEISMIC EVENTS)

- COMPLETE PIPE SEVERANCE PROBABILITY
  WITHOUT SEISMIC EVENTS : 1.68×10^{-12}
  WITH SEISMIC EVENTS : 1.75×10^{-12}

- PERCENT LOCA's THAT ARE SEISMIC INDUCED

$$\frac{1.75-1.68}{1.75} \times 100 = 4\%$$

CLOSING REMARKS

- BASIC MODEL FOR PIPING FAILURE PROBABILITY HAS BEEN CONSTRUCTED.

- MODEL HAS BEEN APPLIED TO CALCULATION OF INFLUENCE OF SEISMIC EVENTS ON INTEGRITY OF MAIN COOLANT PWR PIPING.

- MODEL EASILY EXPANDABLE TO INCLUDE INFLUENCES SUCH AS CRACK INITIATION, BASE METAL, OTHER CRACK GROWTH MECHANISMS, ETC.

- MODEL EASILY APPLICABLE TO OTHER PIPING SYSTEMS AND MATERIALS.

- LARGE UNCERTAINTIES IN SOME INPUTS (SUCH AS INITIAL CRACK DISTRIBUTION).

- THEREFORE, LARGE UNCERTAINTIES IN SOME RESULTS (SUCH AS ABSOLUTE FAILURE PROBABILITIES)

- HOWEVER, NOT SUCH LARGE UNCERTAINTIES IN OTHER <u>RELATIVE</u> RESULTS (SUCH AS PERCENTAGE OF LOCAS THAT ARE SEISMIC INDUCED, OR PREDICTION OF MOST FAILURE-PRONE LOCATIONS).

EIGHTH WATER REACTOR RESEARCH
INFORMATION MEETING

EXPERIMENTAL PROGRAM FOR PIPE-TO-PIPE IMPACT

M. C. C. Bampton
Pacific Northwest Laboratories
Richland, Washington 99352

NRC Technical Monitor: M. Vagins

LICENSING AND SAFETY ISSUE

- Current NRC criteria require the assumption that a whipping pipe shall:
  1) Rupture a pipe of smaller nominal pipe size.
  2) Cause through-wall leakage cracks in pipes of larger size but lesser wall thickness.
  3) Not damage a pipe of greater diameter and equal or greater wall thickness.
- There is a scarcity of analytical and experimental data to substantiate these assumptions.

PROGRAM OBJECTIVES

- Start - August 1980
- Generate an experimental data set of pipe-to-pipe impact results using pipe sizes, impact velocity/energy, support spacing as prime variables.
- Develop functional relationships by analytical models.
- Correlate results with pipe whip code being developed by G. Powell, University of California, Berkeley.

PIPE TO PIPE IMPACT - EXPERIMENTAL PROGRAM

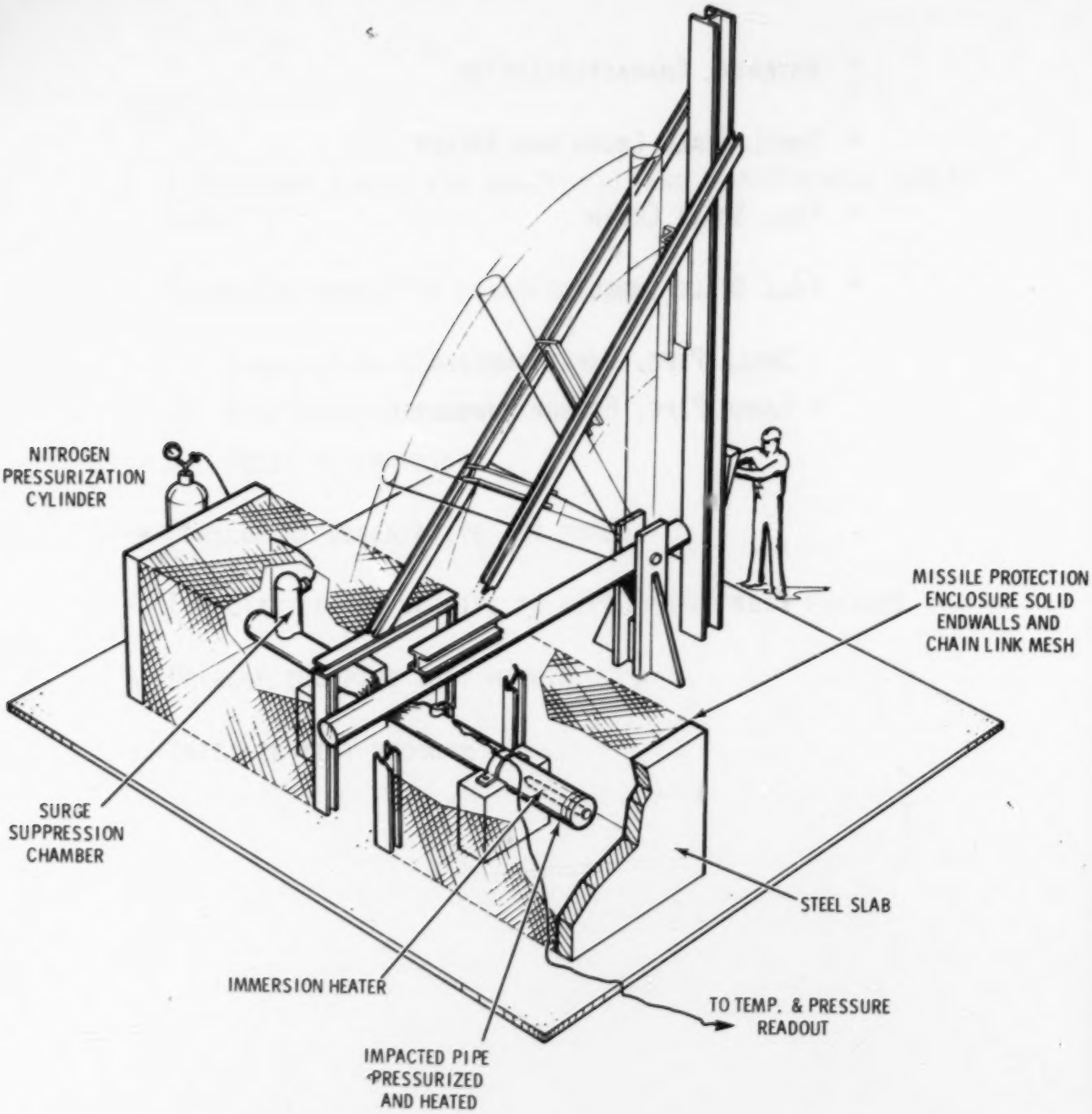
SCOPE:

- LITERATURE SURVEY FOR ANALYTICAL AND EXPERIMENTAL IMPACT DATA.
- ESTABLISH MODELS TO CHARACTERIZE:
  1) LOCAL CRUSH BEHAVIOR
  2) BEAM BENDING DURING IMPACT
  3) ONSET OF CRACKING
- DEVELOP REPRESENTATIVE JET FORCES
- FINALIZE DESIGN OF PIPE TO PIPE IMPACT TEST FIXTURE
- DEVELOP THE PIPE TEST MATRIX
- INITIATE TEST PROGRAM

TEST PROGRAMS

- MATERIAL CHARACTERIZATION
- SMALL SCALE CRUSH AND IMPACT
- FULL SCALE CRUSH
- FULL SCALE IMPACT
  - SMALL PIPE, UNPRESSURIZED
  - LARGE PIPE, ENERGY AUGMENTED

PIPE-TO-PIPE IMPACT TEST FACILITY (CONCEPTUAL DRAWING)



PROTECTION AGAINST PIPE RUPTURE
TWO-PHASE JET LOADS

D. TOMASKO, G. G. WEIGAND, AND S. L. THOMPSON
THERMAL/HYDRAULIC ANALYSIS DIVISION
SANDIA NATIONAL LABORATORIES

PRESENTED AT EIGHTH WATER REACTOR SAFETY
RESEARCH INFORMATION MEETING

GAITHERSBURG, MARYLAND
OCTOBER 31, 1980

PROTECTION AGAINST PIPE RUPTURE

TWO-PHASE JET LOADS

D. Tomasko, G. G. Weigand, S. L. Thompson
Thermal/Hydraulic Analysis Division 4444
Sandia National Laboratories
Albuquerque, NM 87185

In designing a nuclear power plant system, the consequences of a possible pipe break must be mitigated. Snubbers and other forms of pipe restraints must be installed, and jet deflectors must be designed to prevent further effluent damage to the system. Currently, the NRC uses the Moody model for jet load licensing applications. In many cases, the Moody model is overly conservative and can lead to systems which are excessively stiff and difficult to inspect. The Sandia National Laboratories program on two-phase jets is endeavoring to develop a new approximate engineering model which will improve the calculational capability for two-phase jets and result in increased power plant safety.

Initial attempts at model development included using existing computer codes to generate a data base of two-phase jet characteristics and developing an approximate analytic model from first principles. However, both of these approaches have been rejected as either impractical or insolvable. Analytic control volume analysis has been prevented by the pressure and density gradients evident in the two-phase jet, the extreme two-dimensionality of the problem, and the lack of break flow models which have clearly defined exit properties. It was further found that no single existing computer code could successfully model all aspects of a two-phase jet: fluid reservoir, exit piping, choking, and multi-dimensional jet

containment for reasonable computing costs. Subcooled 1-D pipe blowdowns were not well predicted by TRAC-P1A or BEACON/MOD2. TRAC-P1A lacked a containment model and results obtained with the 3-D vessel component were questionable. Although good agreement could be obtained with CSQ when a fine resolution mesh was used, computational costs prevented using CSQ for modeling entire systems except for a small series of tests.

An alternative approach to two-phase jet modeling used coupled computer codes. One code was used to generate the conditions at the rupture site (e.g., RELAP4 and RELAP5) while the other code was used to model the jet and target (SOLA, BEACON/MOD2, and CSQ). Using this approach, the following observations were made: predictions of exit properties using RELAP was difficult to accomplish because of break flow model sensitivity, predictions with a HEM model agreed as well with the experimental data as any other technique, arbitrary boundary conditions could not be used to drive the jet computer code (e.g., a standing shock developed in the CSQ calculation while trying to couple a thermodynamically incompatible boundary condition), non-equilibrium effects were not as important as was fine mesh resolution, pseudo-steady-state calculations for a two-phase jet could be employed because the jet characteristic response times are short compared to the time dependent behavior at the pipe exit, equilibrium methods appear to be sufficient for predicting existing experimental data, higher resolution is required for subcooled flows than for saturated flows, and conservative central impingement pressures could be obtained by assuming a frozen flow model (or cold water analysis).

Currently, we are able to perform jet calculations with CSQ by employing fine resolutions meshes and boundary conditions obtained from a HEM break flow model. However, these calculations execute slowly and are costly. To improve this situation, we are modifying CSQ to execute from 5 to 10 times faster and are writing a special purpose code for steady-state jet analysis using very fast numerical solution techniques which are well suited for this problem. An engineering model will be furnished later this year (FY81) and will be presented in the form of tables and analytic functions.

References

[1] F. J. Moody, "Fluid Reaction and Impingement Loads," Specialty Conference on Structural Design of Nuclear Plant Facilities, Vol. 1, Chicago, IL (December 1973).

[2] RELAP4/MOD6, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems, User's Manual, CDAP TR-003, Idaho National Engineering Laboratory, January 1978.

[3] S. L. Thompson, CSQ-II - An Eulerian Finite Difference Program for Two-Dimensional Material Response, Part I Materials Section, SAND77-1339, Sandia National Laboratories, Albuquerque, NM (January 1979).

[4] C. R. Broadus, S. W. James, W. H. Lee, J. F. Lime, and R. A. Pate, BEACON/MOD2 a CDC 7600 Computer Program for Analyzing The Flow of Mixed Air, Steam, and Water in a Containment System, CDAP-TR-002, EG&G Idaho (December 1977).

[5] Safety Code Development Group, TRAC-P1A: An Advanced Best Estimate Computer Program for PWR LOCA Analysis, LA-7777-MS, NUREG/CR-0665, Vol. 1, Los Alamos National Scientific Laboratory, Los Alamos, NM (May 1979).

[6] R. Eichler, W. Kastner, H. Lochner, K. Riedle, Studies on Critical Two-Phase Flow, NRC-477, U. S. Nuclear Regulatory Commission, Washington, DC (September 1978).

[7] W. Kastner, R. Eichler, K. Riedle, Experimental Studies on Forces Critical Two-Phase Jets for Transverse and Longitudinal Cracks in Pipelines, NRC-478, U. S. Nuclear Regulatory Commission, Washington, DC (September 1978).

[8] LOCA Experiments with a PWR Multi-Compartment Model Containment, BF-RS50-62-5, Battelle Institute E. V. Frankfurt Am Main, 1976.

[9] KFK Nachrichten, Vol. 9, No. 2, NUREG/TR-0038, 1977.

[10] T. F. Kanzleiter, Experimental Investigation of the Jet Force and Its Local Distribution in Case of Rupture of a Primary Circuit Pipe, RS-50-62-7, Frankfurt, Germany Battelle Institut E. V. Frankfurt, Am Main (January 1979).

[11] Abschlussbericht, Forderungsvorhaben BMFT RS 93, Experimentelle Untersuchungen zu Kraften Kritishcher Zweiphasenstrahlen Bei Querund und Langsrissen von Rohrleitungen, RE 23/053/75, Kraftwerk Union.

[12] D. Tomasko, Light Water Reactor Safety Research Program, SAND79-0820, Quarterly Report, Oct-Dec 1978.

[13] D. Tomasko, Light Water Reactor Safety Research Program, Quarterly Report, SAND79-1542, Jan-March 1979.

References Continued

[14] D. Tomasko, Light Water Reactor Safety Research Program, Quarterly Report, SAND79-2057, April-June 1979.

[15] D. Tomasko, "Two-Phase Jet Loads Fiscal Year 1979 Annual Report", NUREG/CR-1292, SAND80-0267.

[16] D. Tomasko, "Two-Phase Jet Loads", Proceedings of the American Nuclear Society, CONF-800403/V1, April 6-8, 1980, Knoxville, TN.

[17] Thermal/Hydraulic Analysis Research Program Quarterly Report, Jan-March 1980, SAND80-1091/1 of 4, NUREG/CR-1157/1 of 4, Sandia National Laboratories, Albuquerque, NM May 1980.

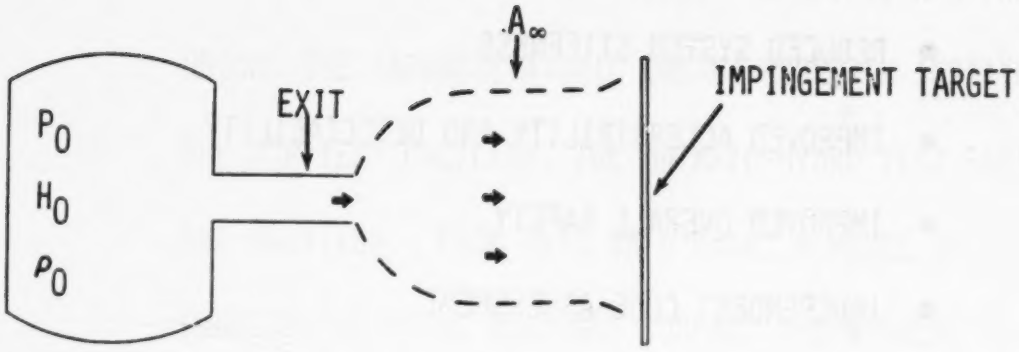
[18] Thermal/Hydraulic Analysis Research Program Quarterly Report, April - June 1980, SAND80-1091/2 of 4, NUREG/CR-1157/2 of 4, Sandia National Laboratories, Albuquerque, NM August 1980.

PROTECTION AGAINST PIPE RUPTURE TWO-PHASE JET LOADS

D. TOMASKO, G. G. WEIGAND, S. L. THOMPSON
THERMAL/HYDRAULIC ANALYSIS DIVISION
SANDIA NATIONAL LABORATORIES

PURPOSE: THE DEVELOPMENT OF AN APPROXIMATE ENGINEERING MODEL TO CHARACTERIZE TWO-PHASE JETS EMANATING FROM BREAKS IN A NUCLEAR REACTOR SYSTEM AND RELATED IMPINGEMENT LOADS.

MOODY MODEL
CURRENTLY USED FOR LICENSING APPLICATIONS



APPROXIMATIONS USED IN MOODY MODEL:

- CIRCUMFERENTIAL BREAK MODELED IN 1-D GEOMETRY
- ASYMPTOTIC JET EXPANSION
- THERMODYNAMIC EQUILIBRIUM ASSUMED
- IMMEDIATE SONIC DISCHARGE ASSUMED AT THE PIPE EXIT
- EQUAL RADIAL AND AXIAL VELOCITIES
- FLAT TARGET USED FOR LOADING
- MOODY BREAK FLOW MODEL

BENEFITS OF NEW MODEL

- IMPROVED CALCULATIONAL CAPABILITY FOR TWO-PHASE JETS AND LOADS
- REDUCED CONSERVATISM
- REDUCED COST OF SAFETY SYSTEMS
- REDUCED SYSTEM STIFFNESS
- IMPROVED ACCESSIBILITY AND DETECTABILITY
- IMPROVED OVERALL SAFETY
- INDEPENDENT CODE ASSESSMENT

APPROACH TO MODEL DEVELOPMENT AS ENVISIONED AT START OF PROJECT

PATH ONE: GENERATE A DATA BASE OF TWO-PHASE JET CHARACTERISTICS USING EXISTING COMPUTER CODES (TRAC-P1A, BEACON/MOD2, CSQ, ETC.) WHICH HAVE BEEN ASSESSED AGAINST EXPERIMENTAL DATA FROM BATTELLE-FRANKFURT, KRAFTWERK UNION, THE JAPANESE ATOMIC ENERGY RESEARCH INSTITUTE, THE HDR TEST FACILITY, THE ONTARIO-HYDRO TEST FACILITY, AND MARVIKEN. FROM THIS DATA BASE, A MODEL IS CONSTRUCTED USING RESPONSE SURFACE TECHNIQUES.

PATH TWO: DEVELOP AN APPROXIMATE ANALYTIC MODEL OF TWO-PHASE JET CHARACTERISTICS FROM FIRST PRINCIPLES. THE MODEL IS ASSESSED IN THE SAME MANNER AS ABOVE.

RESULTS FOR INITIAL PROJECT APPROACHES

- BOTH APPROACHES REJECTED AS IMPRACTICAL

OUTLINE FOR REMAINDER OF DISCUSSION

- DIFFICULTIES WITH THE ANALYTIC APPROACH
- DIFFICULTIES WITH COMPUTER CODE APPROACH
- OUTLINE OF PRACTICAL APPROACH BEING FOLLOWED TO COMPLETE PROJECT

ANALYTICAL MODEL DEVELOPMENT

ORIGINAL OBJECTIVE

- SIMPLE EXTENSION OF MOODY'S ASYMPTOTIC MODEL
- CALCULATE THE TWO-PHASE JET'S BEHAVIOR FOR OTHER THAN THE ASYMPTOTIC LIMIT

ANALYTICAL APPROACH

- CONTROL VOLUME ANALYSIS WITH TWO FLUID STREAMS (LIQUID AND VAPOR) WHICH ARE NOT NECESSARILY IN EQUILIBRIUM
- AERODYNAMIC, MASS TRANSFER, AND HEAT TRANSFER MODELING

PROBLEMS WITH ANALYTICAL MODEL DEVELOPMENT

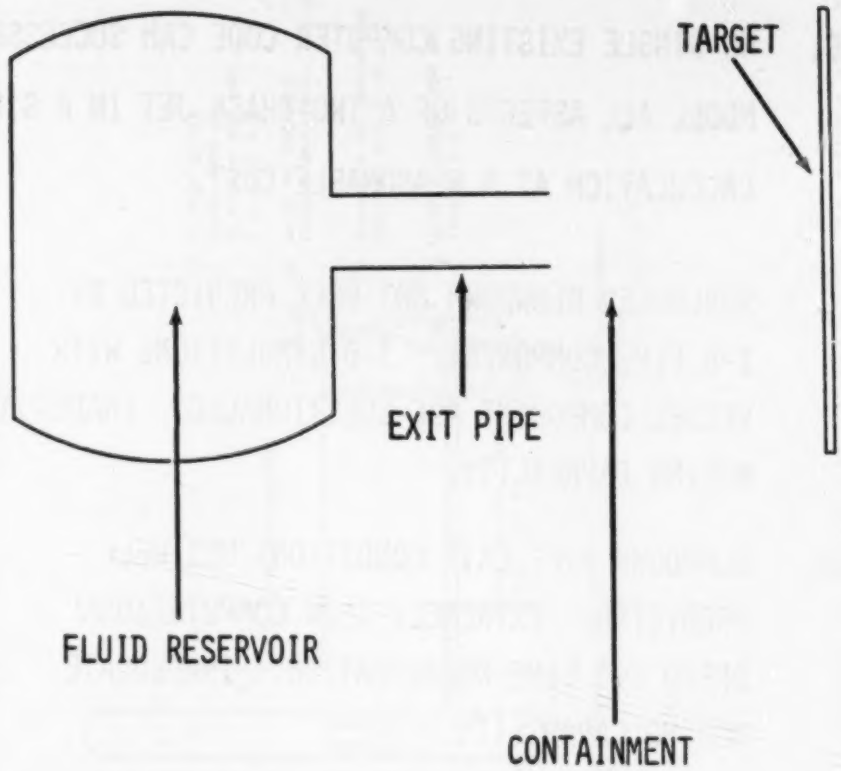
CONTROL VOLUME MODEL IS INADEQUATE FOR TWO-PHASE JET MODELING

- MASS TRANSFER RATES DRIVE SOLUTION NONPHYSICAL FOR LOW QUALITY FLOWS
- CONTROL VOLUME MODELING PRESUPPOSES A TWO-PHASE JET BEHAVIOR (SMALL LATERAL PRESSURE GRADIENTS) AND SHAPE WHICH ARE UNLIKE THE TRUE JET

USEFUL RESULTS FROM THE ANALYTICAL MODEL DEVELOPMENT EFFORT

- 2-D AND 3-D FLOW EFFECTS ARE IMPORTANT
- HIGH MASS TRANSFER RATES ARE TO BE EXPECTED NEAR THE BREAK FOR SUBCOOLED FLOWS (MAY INDICATE THAT IN THE TWO-PHASE JET THE DEPARTURE FROM EQUILIBRIUM FLOW IS NEGLIGIBLE)
- BREAK FLOW MODELS WHICH DO NOT HAVE CLEARLY DEFINED EXIT PROPERTIES BECAUSE OF EMPIRICAL CORRELATIONS OR MULTIPLIERS ARE INADEQUATE FOR INITIALIZING CONTROL VOLUME OR MULTI-DIMENSIONAL JET CALCULATIONS

SINGLE-CODE CONCEPT FOR MODEL DEVELOPMENT



ENTIRE SYSTEM IS MODELED BY A SINGLE COMPUTER CODE INCLUDING:

- FLUID RESERVOIR
- EXIT PIPE
- CONTAINMENT
- TARGET

SINGLE-CODE CONCEPT

CONCLUSIONS: NO SINGLE EXISTING COMPUTER CODE CAN SUCCESSFULLY MODEL ALL ASPECTS OF A TWO-PHASE JET IN A SINGLE CALCULATION AT A REASONABLE COST.

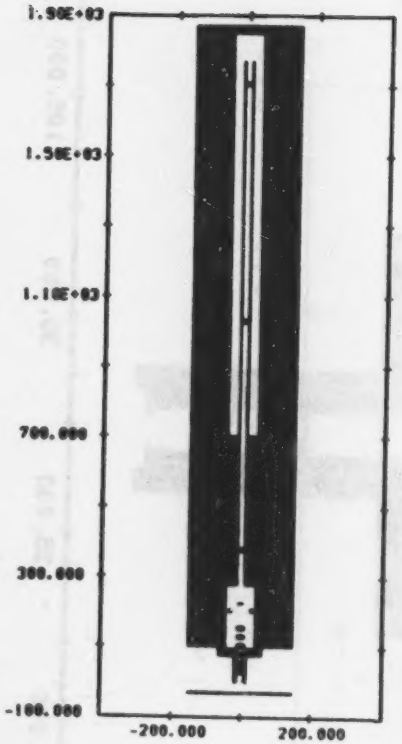
TRAC-P1A — SUBCOOLED BLOWDOWN NOT WELL PREDICTED BY 1-D PIPE COMPONENT. 3-D SIMULATIONS WITH VESSEL COMPONENT ARE QUESTIONALBE. INADEQUATE NODING CAPABILITY.

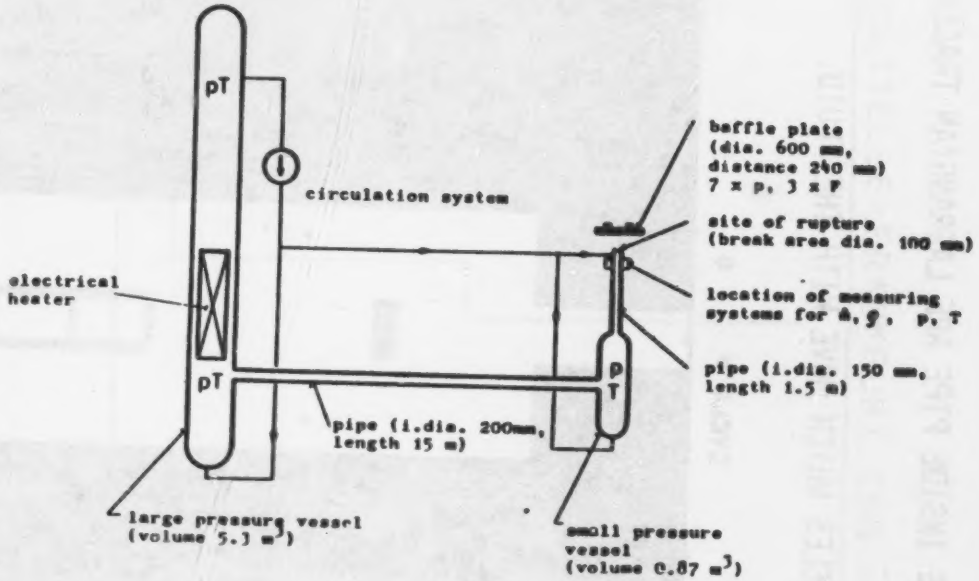
BEACON/MOD2 — BLOWDOWN PIPE EXIT CONDITIONS NOT WELL PREDICTED. EXTREMELY SLOW COMPUTATIONAL SPEED FOR FINE NODALIZATION. INADEQUATE NODING CAPABILITY.

CSQ — CAN SUCCESSFULLY MODEL ENTIRE SYSTEM, BUT COMPUTATIONAL COSTS ARE PROHIBITIVE. EXTREMELY FINE RESOLUTION REQUIRED NEAR PIPE EXIT AND TARGET.

TWO-DIMENSIONAL CALCULATIONS REQUIRED IN PIPE FOR BOTH BREAK FLOW AND JET FORMATION.

CSQ MODEL OF BATTELLE-FRANKFURT RS-50 EXPERIMENTS C10, C11 AND C12. CSQ DIMENSIONS SHOWN IN CM. THE NOZZLE AND IMPINGEMENT PLATE ARE AT BOTTOM.



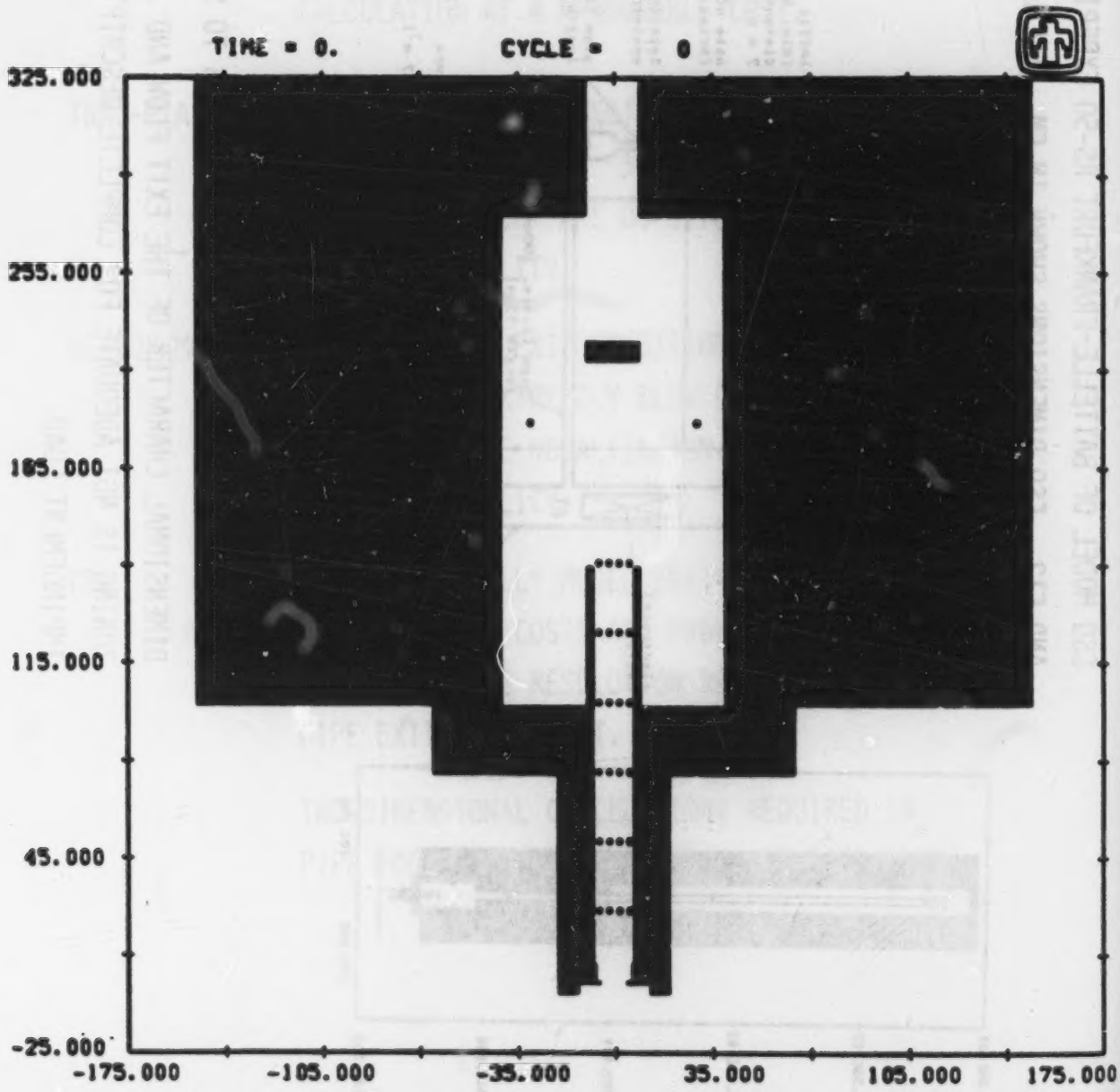


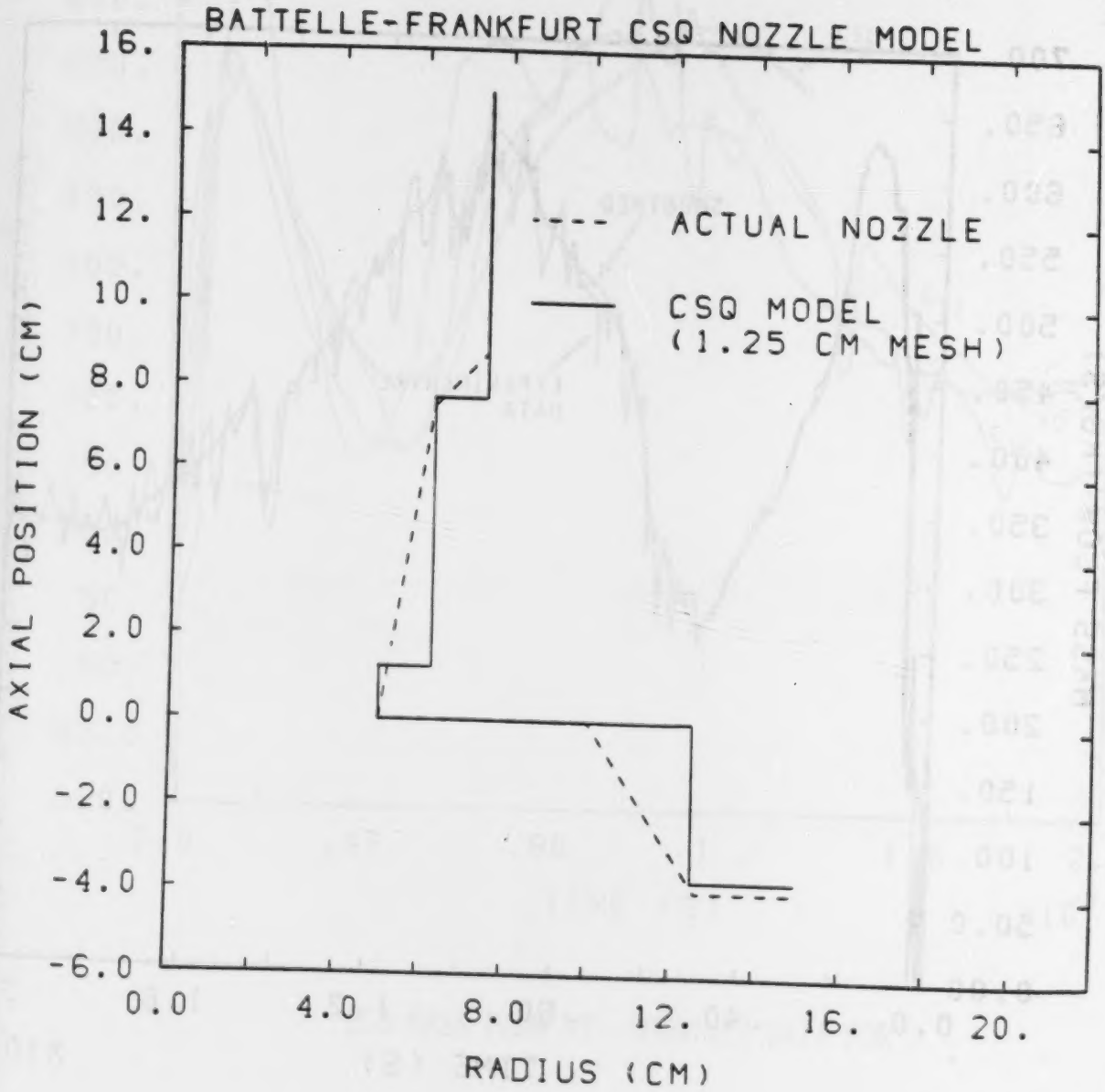
THE MAIN PURPOSE OF THIS CALCULATION IS TO STUDY THE TWO-DIMENSIONAL CHARACTER OF THE EXIT FLOW AND JET FORMATION. ZONING IS NOT ADEQUATE FOR COMPLETE DESCRIPTION OF THE IMPINGEMENT LOAD.

CSQ MODEL OF SMALL PRESSURE VESSEL.

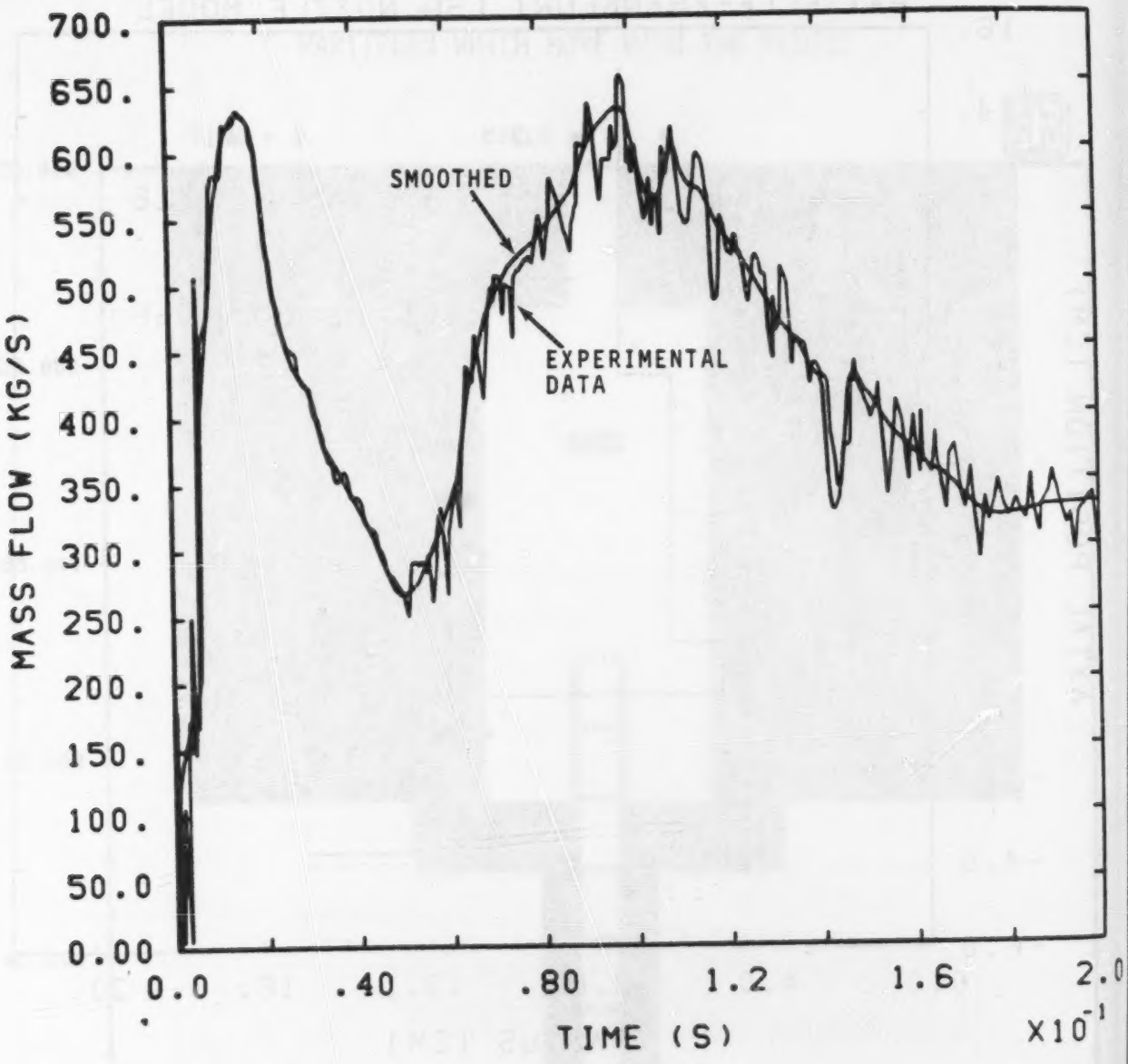
POINTS INSIDE PIPE ARE LAGRANGIAN TRACER

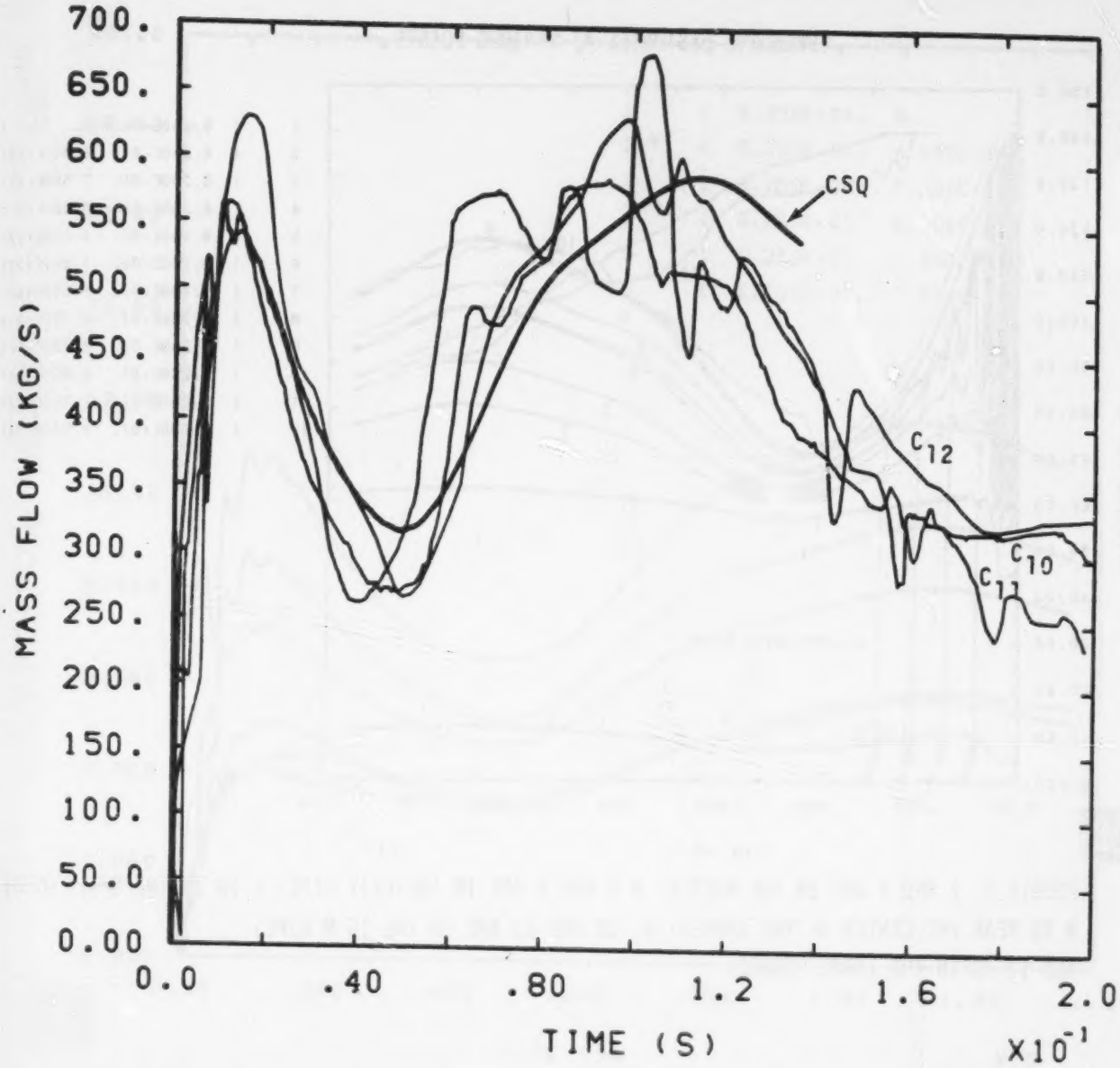
PARTICLES WHICH MOVE WITH THE FLUID.




BATTELLE-FRANKFURT CSQ NOZZLE MODEL
ACTUAL NOZZLE
CSQ MODEL
(1.25 CM MESH)
AXIAL POSITION (CM)
RADIUS (CM)
16.
14.
12.
10.
8.0
6.0
4.0
2.0
0.0
-2.0
-4.0
-6.0
0.0
4.0
8.0
12.
16.
20.

SMOOTHED VS. EXPERIMENTAL DATA FOR EXPERIMENT RS-50-C12.

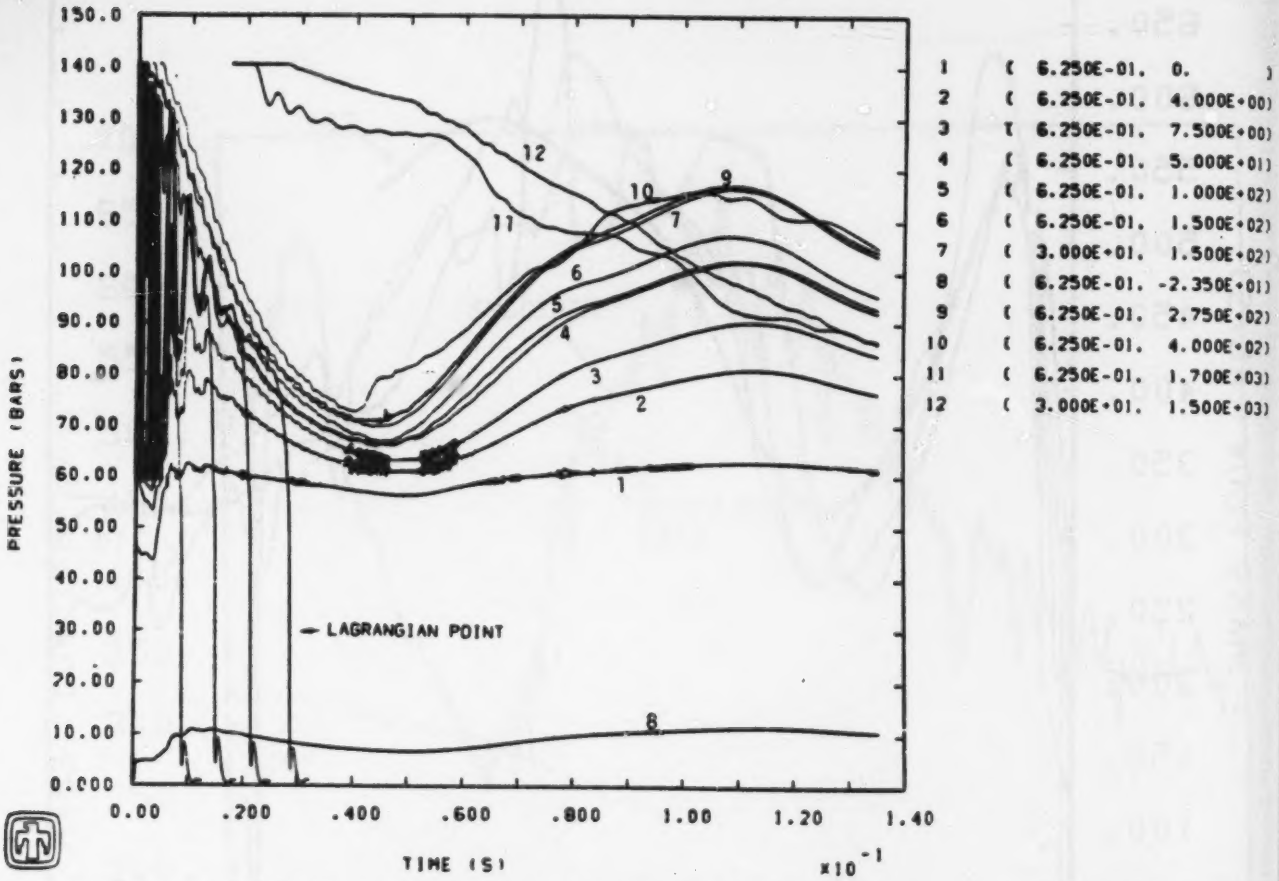




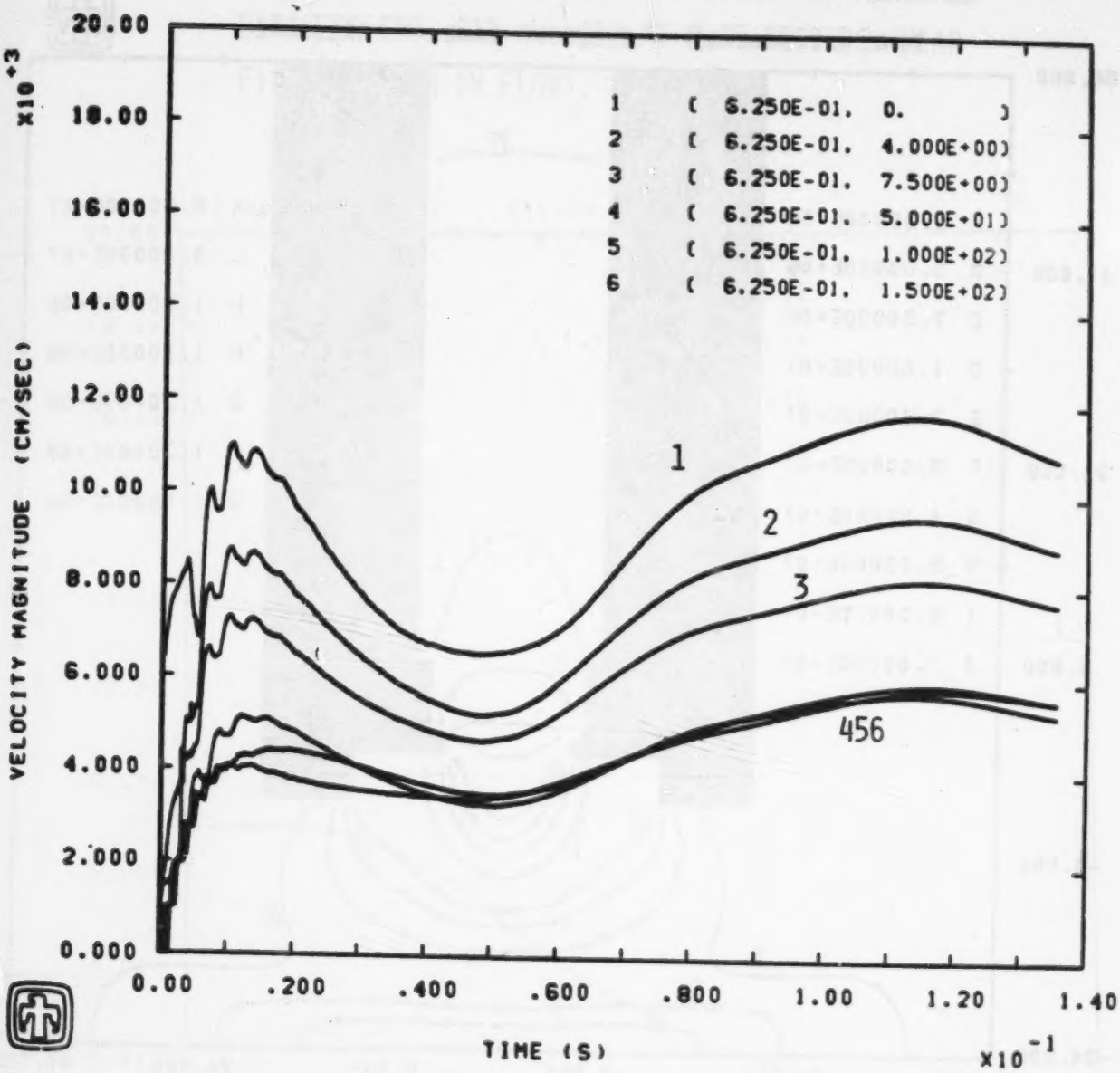
CSQ MASS FLOW VS. SMOOTHED DATA FOR

RS-50-C10, C11, C12.

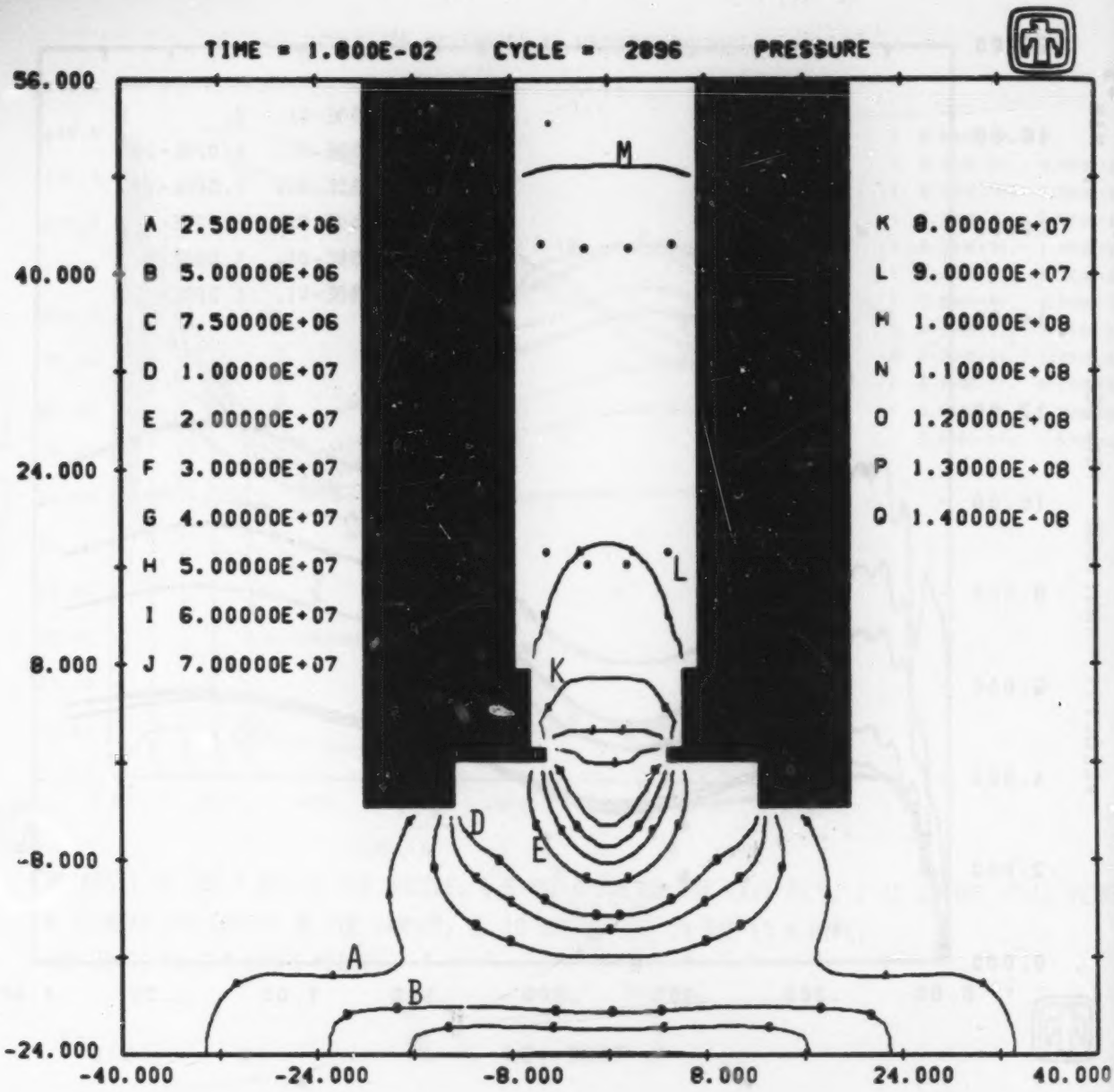
PRESSURE HISTORIES AT VARIOUS POINTS.



POINTS 1, 2 AND 3 ARE IN THE NOZZLE; 4 5 AND 6 ARE IN THE EXIT PIPE; 7 IS IN THE SMALL VESSEL; 8 IS NEAR THE CENTER OF THE TARGET; 9, 10 AND 11 ARE IN THE 15 M PIPE; AND 12 IS IN THE LARGE VESSEL.



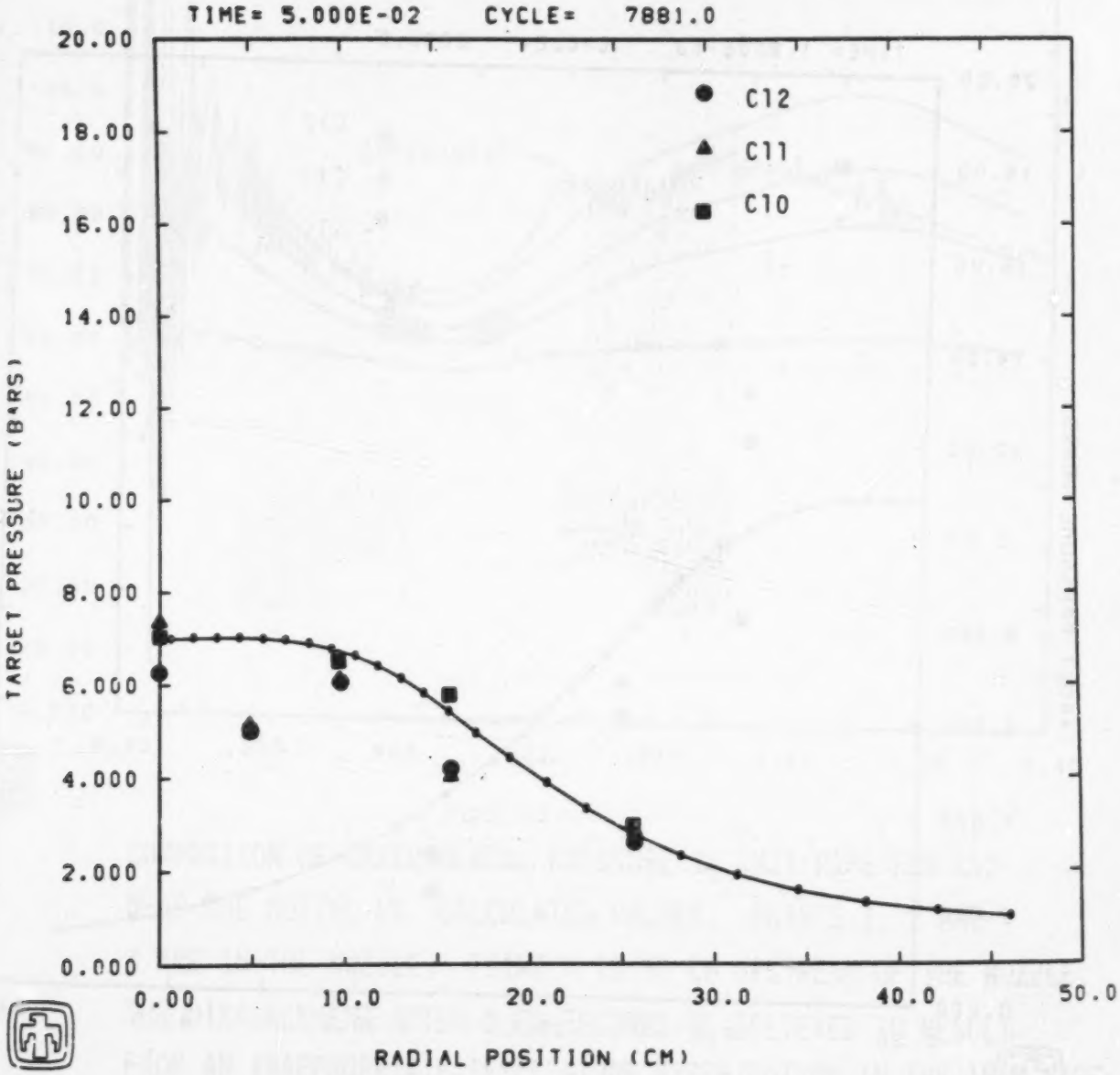
VELOCITY HISTORIES AT VARIOUS POINTS IN THE NOZZLE AND EXIT PIPE.



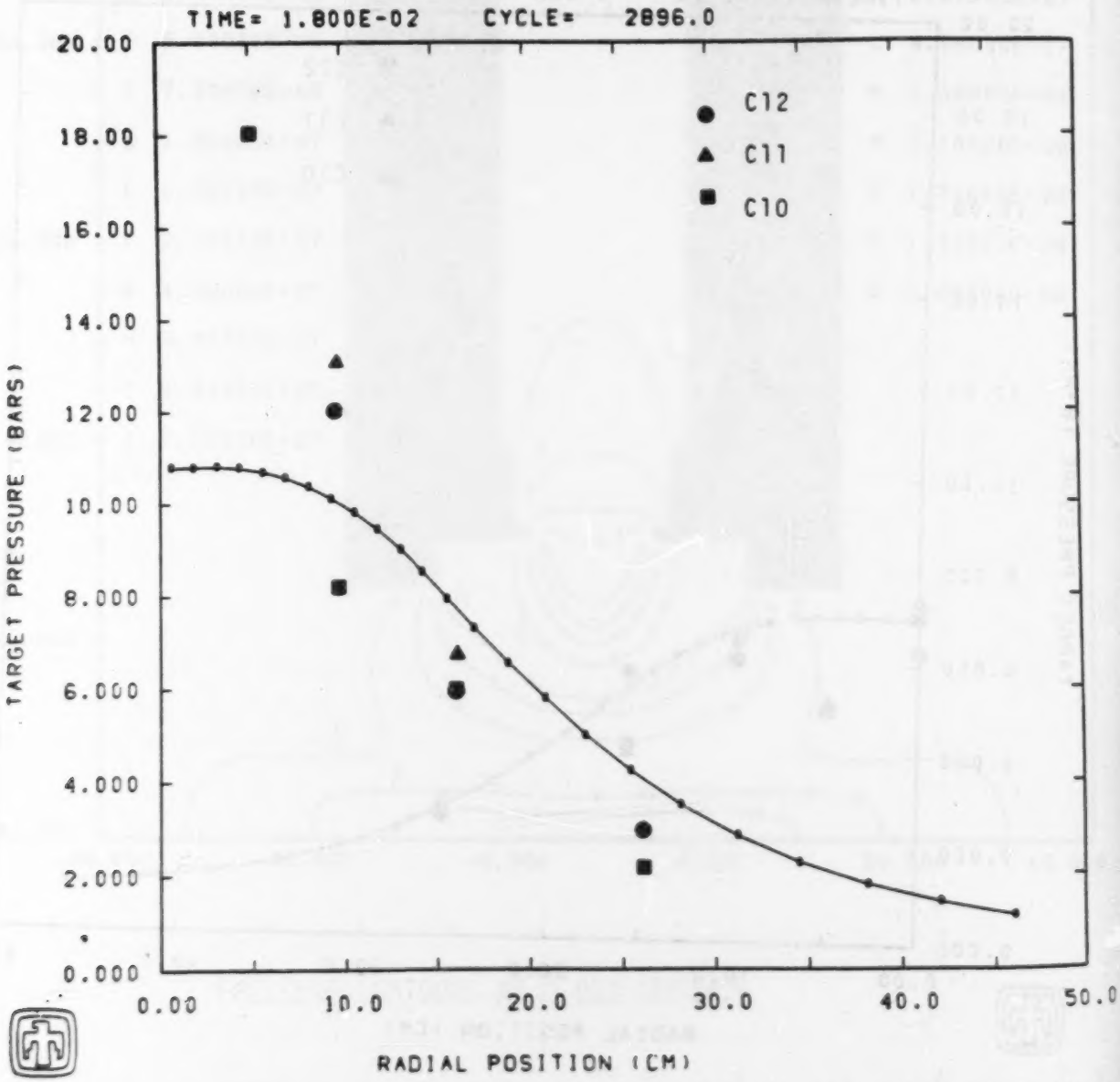
PRESSURE CONTOURS AT 0.018 SECONDS.

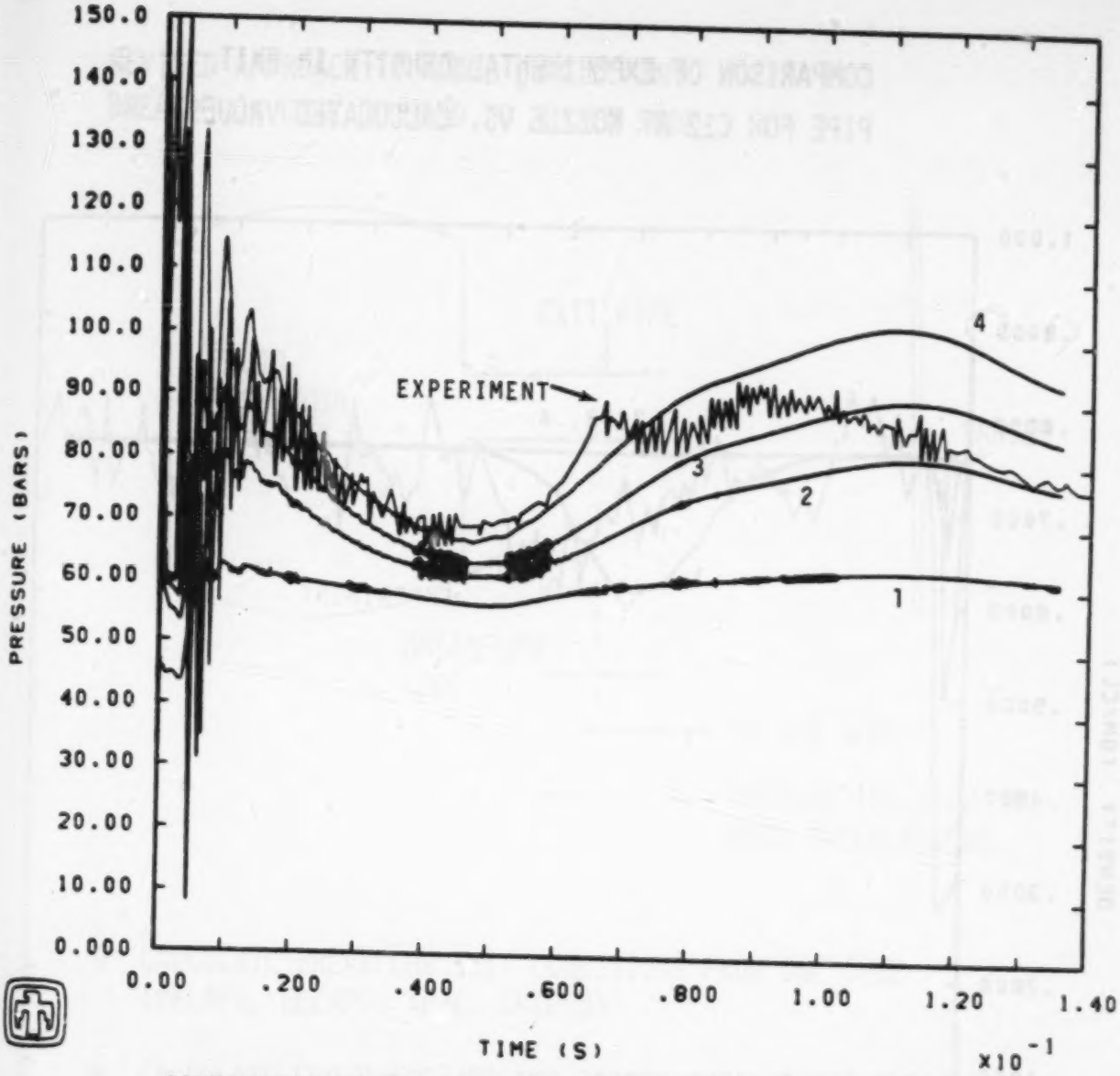
PRESSURE VALUES SHOWN IN DYNES/CM^2 (1 BAR = 10^6 DYNES/CM^2)

CSQ CALCULATED IMPINGEMENT PRESSURE VS. MEASURED DATA FOR C10, C11 AND C12 AT 0.05 SECONDS (NEAR FIRST MINIMUM IN FLOW).



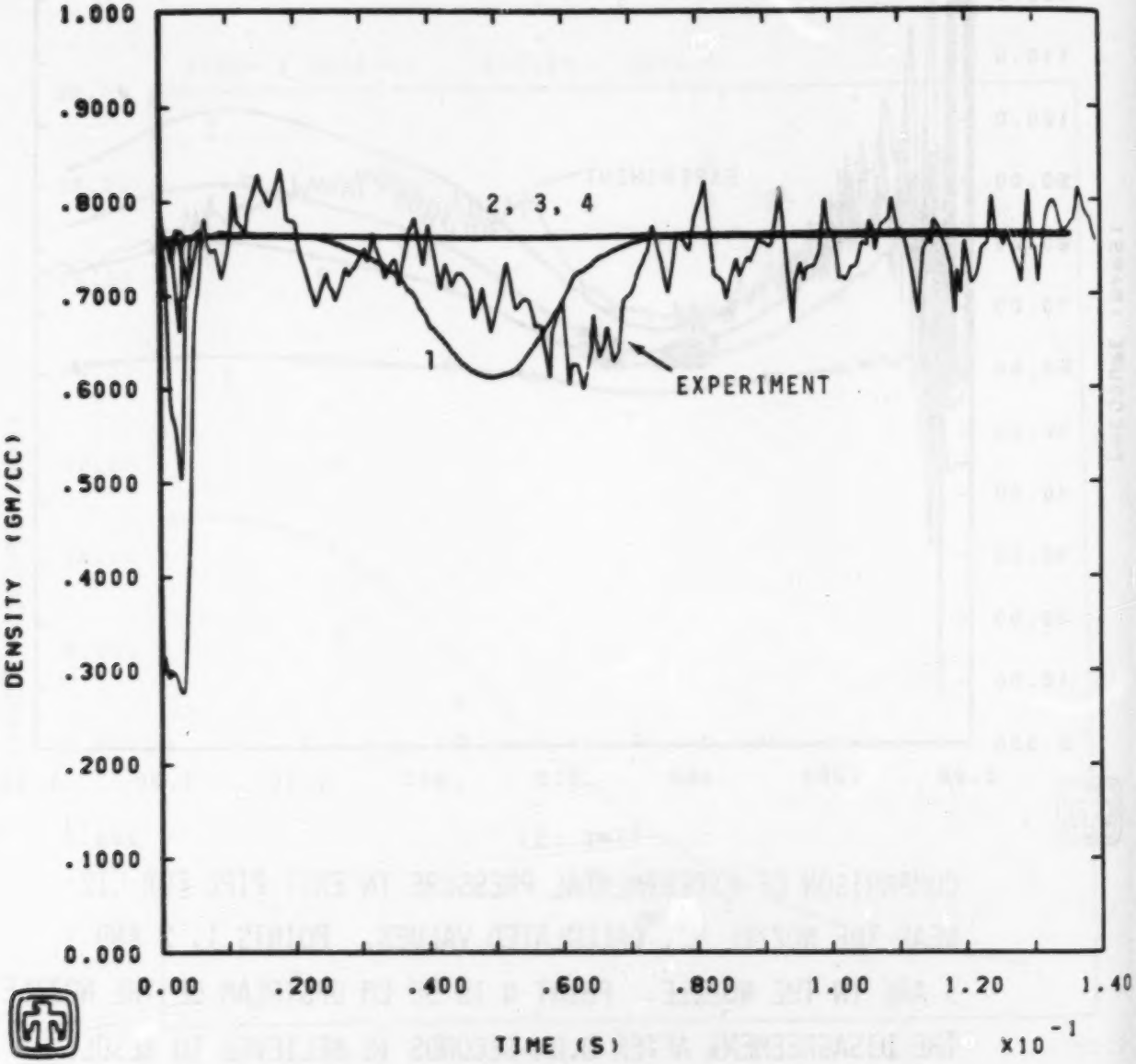
CSQ CALCULATED IMPINGEMENT PRESSURE VS. MEASURED DATA FOR C12, C11 AND C12 AT 0.018 SECONDS (NEAR FIRST PEAK IN FLOW). INSUFFICIENT NODALIZATION NEAR NOZZLE EXIT RESULTS IN POOR PREDICTION.





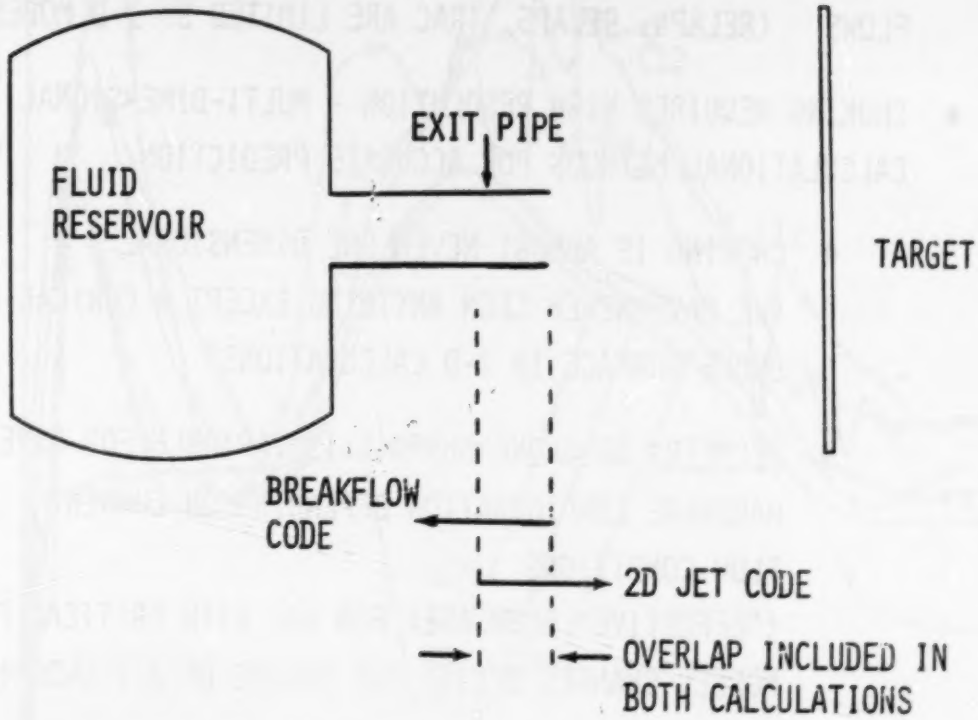
COMPARISON OF EXPERIMENTAL PRESSURE IN EXIT PIPE FOR C12 NEAR THE NOZZEL VS. CALCULATED VALUES. POINTS 1, 2 AND 3 ARE IN THE NOZZLE. POINT 4 IS 50 CM UPSTREAM OF THE NOZZLE. THE DISAGREEMENT AFTER 0.04 SECONDS IS BELIEVED TO RESULT FROM AN INAPPROPRIATE TEMPERATURE DISTRIBUTION IN THE 15 M PIPE AND INCORRECT SHAPE OF THE LARGE VESSEL.

COMPARISON OF EXPERIMENTAL DENSITY IN EXIT PIPE FOR C12 AT NOZZLE VS. CALCULATED VALUES.



POINT 1 IS AT THE NOZZLE EXIT PLANE. POINTS 2, 3 AND 4 ARE UPSTREAM.

REVISED APPROACH TO MODEL DEVELOPMENT
BREAK-FLOW CODE COUPLED TO 2D JET CODE

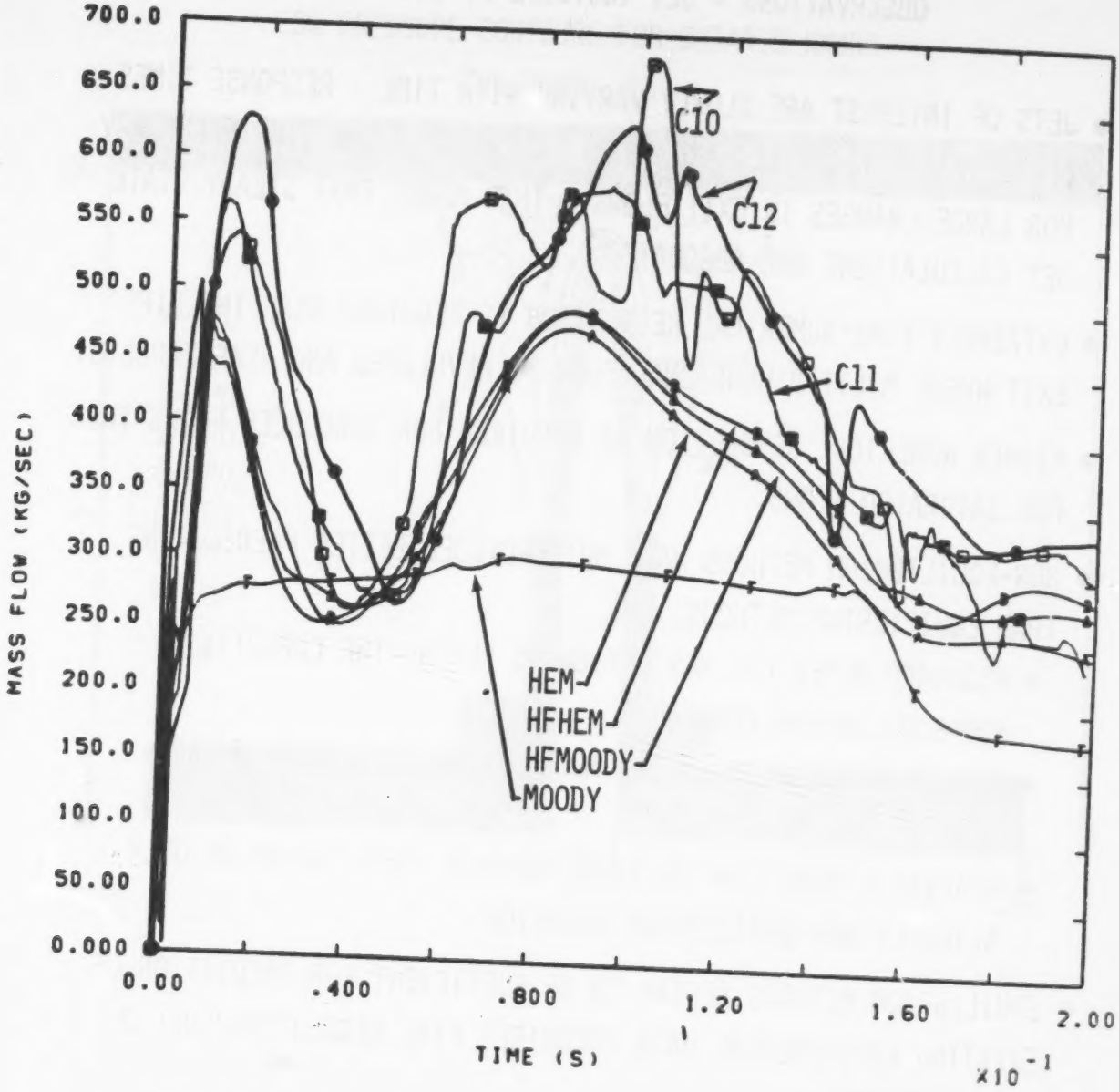


- CALCULATE BREAKFLOW EXIT CONDITIONS FROM ONE CODE (RELAP4, RELAP5, TRAC, CRTFLO)

- CALCULATE TWO PHASE JET AND TARGET IMPINGEMENT PRESSURES WITH ANOTHER CODE USING EXIT OR SLIGHTLY UPSTREAM CONDITIONS AS INPUT BOUNDARY CONDITIONS. (BEACON/MOD2, SOLA, CSQ, SSJET)

OBSERVATIONS - BREAK FLOW

- REACTOR BASED SYSTEM CODES CANNOT RELIABLY PREDICT BREAK FLOWS (RELAP4, RELAP5, TRAC ARE LIMITED BY 1-D MODELS)
- CHOKING REQUIRES HIGH RESOLUTION - MULTI-DIMENSIONAL CALCULATIONAL METHODS FOR ACCURATE PREDICTION
    - CHOKING IS ALMOST NEVER ONE DIMENSIONAL (WE HAVE NEVER SEEN ANYTHING EXCEPT A CONICAL CHOKE SURFACE IN 2-D CALCULATIONS)
    - GEOMETRY OF CHOKE SURFACE IS VARIABLE FOR GIVEN HARDWARE CONFIGURATION DEPENDING ON CURRENT FLOW CONDITIONS
      ("EFFECTIVE" FLOW AREA FOR USE WITH CRITICAL FLOW MODELS CHANGES DURING THE COURSE OF A BLOWDOWN)
    - EQUILIBRIUM MODELS IN 2-D CALCULATIONS DO AS WELL AS ANY MODEL IN MATCHING EXPERIMENTAL DATA

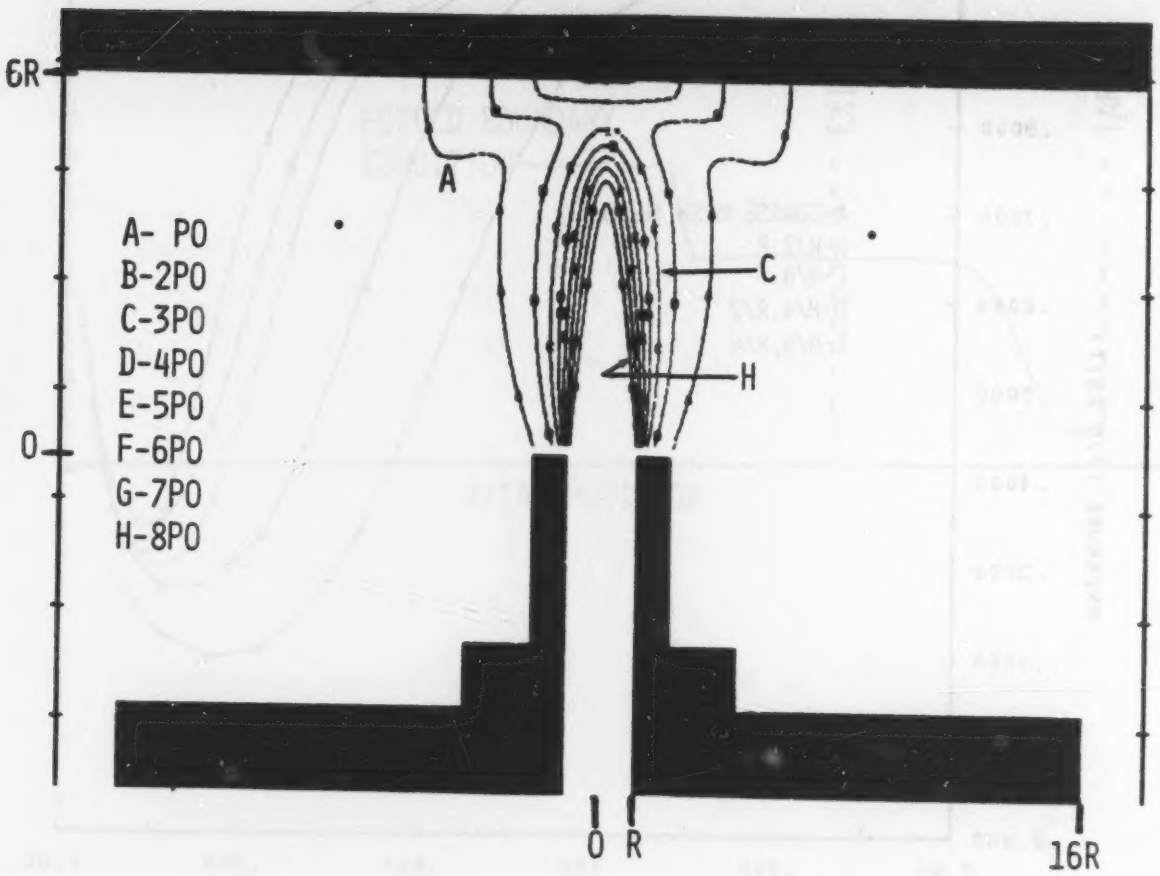


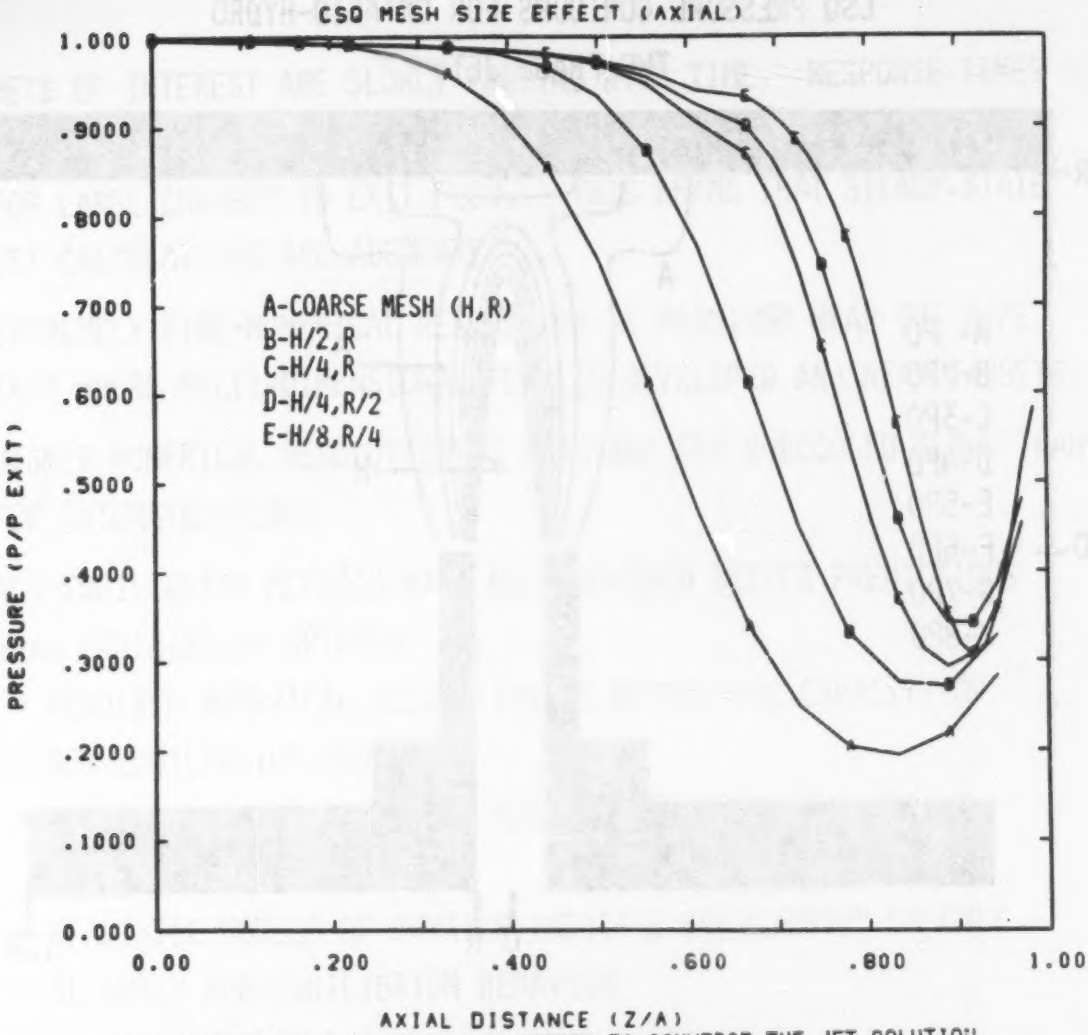
RELAP4/MOD6 EXIT MASS FLOWS WITH INDICATED CRITICAL FLOW MODELS VS SMOOTHED EXPERIMENTAL DATA FOR RS-50-C10, C11 AND C12.

OBSERVATIONS - JET (OUTSIDE PIPING)

- JETS OF INTEREST ARE SLOWLY VARYING WITH TIME. RESPONSE TIMES OF JET CHARACTERISTICS ARE SHORT COMPARED TO THE TIME NECESSARY FOR LARGE CHANGES IN EXIT FLOWS. THIS MEANS THAT STEADY-STATE JET CALCULATIONS ARE ADEQUATE.
- EXTREMELY FINE NUMERICAL RESOLUTION IS REQUIRED NEAR THE PIPE EXIT WHERE MULTI-DIMENSIONAL FLOW IS DEVELOPED AND NEAR TARGETS.
- HIGHER NUMERICAL RESOLUTION IS REQUIRED FOR SUBCOOLED FLOWS THAN FOR SATURATED FLOWS.
- NON-EQUILIBRIUM METHODS HAVE NOT PROVIDED BETTER PREDICTIONS THAN EQUILIBRIUM METHODS.
  - REQUIRED NUMERICAL RESOLUTION IS BEYOND THE CAPACITY OF NON-EQUILIBRIUM CODES.
  - NON-EQUILIBRIUM CODES ARE MUCH SLOWER THAN EQUILIBRIUM CODES.
  - REALISTIC MODELS OF PHYSICS INDICATE EQUILIBRIUM OR ONLY SLIGHTLY NON-EQUILIBRIUM BEHAVIOR.
- EQUILIBRIUM METHODS APPEAR TO BE SUFFICIENT FOR PREDICTION OF EXISTING EXPERIMENTAL DATA (REQUIRES FINE RESOLUTION/FULL 2- OR 3-D TREATMENT).
- MODELS IN BOTH BREAK FLOW CODES AND JET CODES MUST BE THERMODYNAMICALLY COMPATIBLE.
- CONSERVATIVE CENTRAL IMPINGEMENT PRESSURES CAN BE OBTAINED FROM FROZEN FLOW OR COLD WATER CALCULATIONS FOR SUBCOOLED CONDITIONS.

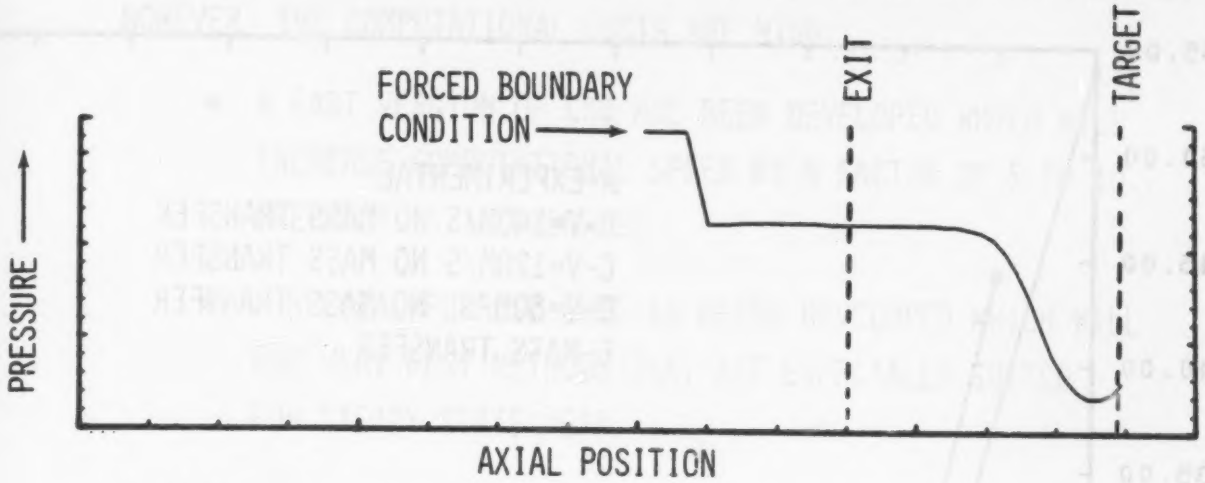
CSQ PRESSURE CONTOURS FOR ONTARIO-HYDRO
TWO-PHASE JET





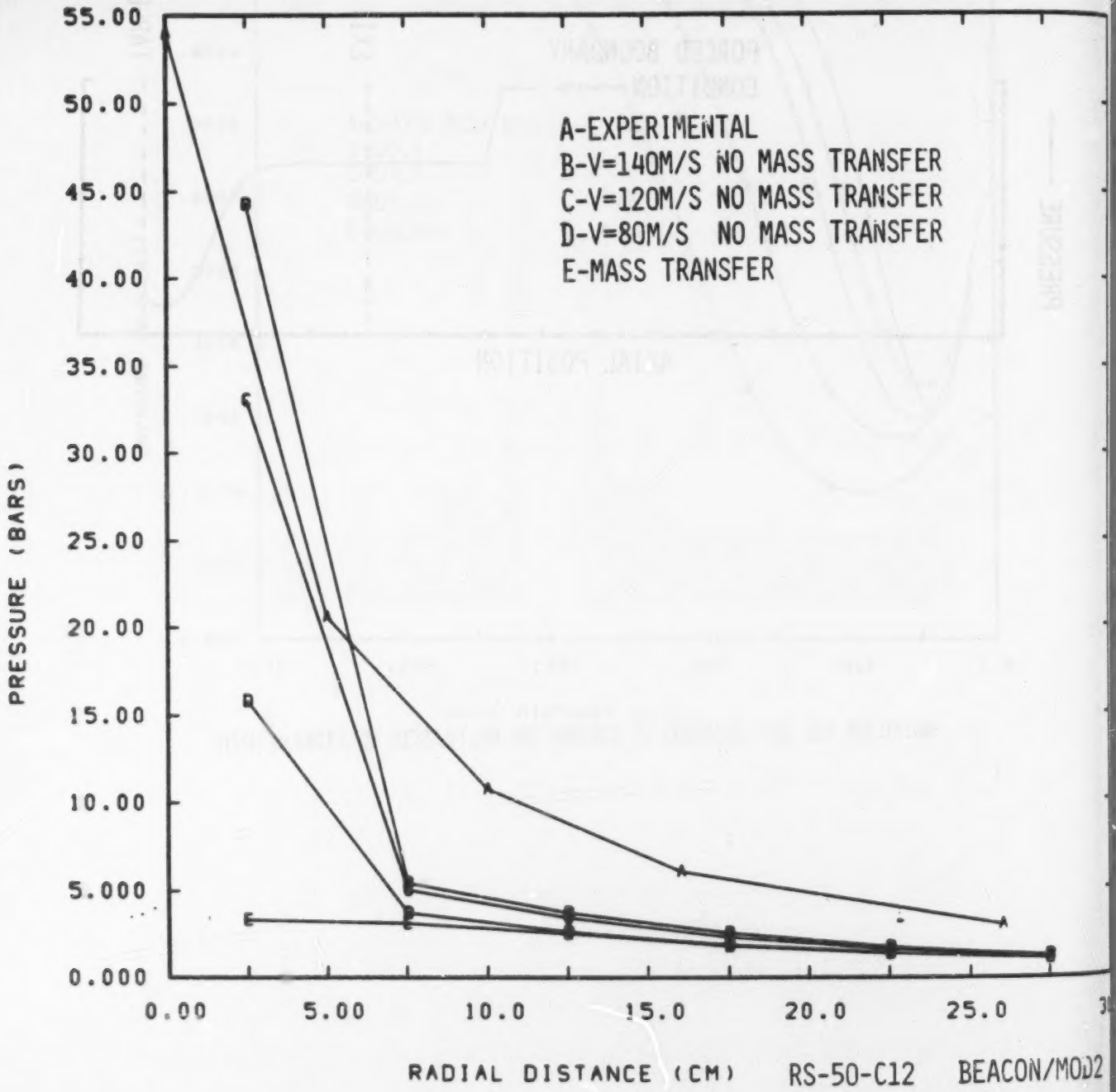
HIGH DEGREES OF RESOLUTION ARE NEEDED TO CONVERGE THE JET SOLUTION

ARBITRARY BOUNDARY CONDITIONS CAN NOT BE COUPLED FROM ONE CODE TO ANOTHER.



FOR SUBCOOLED STAGNATION CONDITIONS, CONSERVATIVE RESULTS COULD BE OBTAINED BY ASSUMING A FROZEN-FLOW MODEL ON COLD WATER IN THE JET.

BF RS-50-C12 COMPARISONS BEACON/MOD2



CONCLUSIONS AND PLANS

- WE CAN NOW PRODUCE IMPINGEMENT LOADS FOR 2-D SYMMETRIC SYSTEMS. HOWEVER, THE COMPUTATIONAL COSTS ARE HIGH.
    - A FAST VERSION OF CSQ HAS BEEN DEVELOPED WHICH WILL INCREASE COMPUTATIONAL SPEED BY A FACTOR OF 5 TO 10. CHECKOUT IS IN PROGRESS.
    - A SPECIAL PURPOSE CODE IS BEING DEVELOPED WHICH WILL USE VERY FAST METHODS THAT ARE ESPECIALLY SUITED FOR STEADY STATE JETS.
- AN ENGINEERING MODEL (TABLES AND ANALYTIC FUNCTIONS FOR 2-D SYSTEMS) WILL BE AVAILABLE IN FY81.
- EXTENSION TO 3-D SITUATIONS IS UNLIKELY IN FY81.
    - COMPUTATIONAL COSTS ARE EXTREMELY HIGH.
    - NO EXPERIMENTAL DATA FOR 3-D LEAKS OR CRACKS.

PNL-SA-9022

Progress and Plans for Steam Generator Integrity Research

PWR Steam Generator Tube Integrity Program

R. A. Clark, Project Manager
V. F. FitzPatrick, Deputy Project Manager

October 1980

Presented at the Eighth Water Reactor Safety Research Information Meeting
Gaithersburg, Maryland
October 23-31, 1980

Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
by Battelle Memorial Institute

Battelle

PNL-SA-9022

PNL-SA-9022

PROGRESS AND PLANS FOR STEAM GENERATOR INTEGRITY RESEARCH

PWR STEAM GENERATOR TUBE INTEGRITY PROGRAM

R. A. Clark, Project Manager
V. F. FitzPatrick, Deputy Project Manager

October 1980

Presented at the Eighth Water Reactor Safety
Research Information Meeting
Gaithersburg, Maryland
October 23-31, 1980

Work Supported by
the U. S. Nuclear Regulatory Commission
NRC FIN No. B-2097
Dr. Joseph Muscara, Program Manager

Pacific Northwest Laboratory
Richland, Washington 99352

STEAM GENERATOR TUBE INTEGRITY PROGRAM

The Steam Generator Tube Integrity Program is part of NRC research efforts concerning primary system integrity. Under this program mathematical models have been developed which relate defect geometry and severity in Inconel 600 steam generator tubes to remaining service strength. The models were developed based on experimental data derived from burst and collapse tests at typical operating temperatures of steam generator tubes with mechanically or chemically placed defects. The types of artificial defects were chosen to simulate known or postulated service degradation. These defects included uniform circumferential thinning (a general wastage), elliptical wastage (loss of metal from one area), electro discharge machined (EDM) notches or chemically produced stress corrosion cracks, dented configurations, and combinations of the above. The test matrix included wide variations in the size (e.g. length) of defects as well as severity (expressed as percentage of tube wall).

Coupled with the mechanical testing aspects of the program is an extensive nondestructive testing (NDT) effort. Defect characterization has employed a variety of methods, but mainly single and multifrequency eddy current techniques. Positive sizing of artifically produced defects, through replicating techniques has allowed an evaluation of the accuracy of NDT measurements. Joining the NDT information with the models derived from burst and collapse test data has allowed development of predictive curves which relate typical inservice inspection (ISI) eddy current data to remaining service strength of defected tubes. These predictive curves have verified, to date, that NRC mandated tube plugging criteria are conservative.

Work to date has all been conducted on laboratory simulations of postulated or known service defects. This work while demonstrating the general conservative nature of tube plugging criteria, requires verification on actual service degraded components. It remains to be established that the laboratory produced defects from which our tube integrity models were derived, are in fact valid simulations of service degradation. It must also be verified that some undetected or unidentified defect or defect combination does not exist in the generator. These and several other issues, as discussed below, can only be addressed through access to extensive amounts of removed from service steam generator tubing. To provide this out of service material the program has acquired and shipped to the Hanford Reservation (Hanford, Washington) a nuclear steam generator removed from service at the Surry Nuclear Station, operated by Virginia Electric and Power Company. Details of the acquisition and transport of this generator are covered in the Program Accomplishments section of this paper. Research plans for the generator are then detailed in the following section.

PROGRAM ACCOMPLISHMENTS DURING FY 1980

The major program accomplishment involved licensing and transport activities to acquire and move a retired from service PWR steam generator to the Hanford Reservation in Washington State. The "A" steam generator, removed from the Surry II Nuclear Power Station, Surry, Virginia, in January, 1979 after slightly less than six years service, was successfully transported to Hanford between April 20 and May 25, 1980. Preceeding this move was over a year long effort to acquire necessary licensing approvals and establish a series of subcontracts that needed to closely intermesh. Issues that were resolved prior to shipping included ownership, point of ownership transfer, source of Price Anderson Act insurance indemnification, and government agency jurisdiction areas. The shipment was conducted under the auspices of the Department of Transportation, because the Department of Energy took possession of the generator at the Surry site. The inventory of radioactive material in the generator was mixed corrosion products, principally Co^{58} and Co^{60}. Based on radiation readings taken at the time the generator exited the Surry II containment building and analysis of the corrosion products in the channel head it was estimated that the generator contained 120 to 180 curies of radioactive inventory at the time of shipment. This radioactivity was almost entirely bound up in metal oxides on the primary side of the steam generator tubes, and in the channel head. Attempts were made to license the generator as a low specific activity (LSA) radioactive shipment. Calculations based on radioactive inventory per weight of steam generator tubing, according to requirements of 49 CFR 173.392, used as justification for LSA shipment, were not approved by the licensing agency. Rather the calculation required was radioactivity per unit weight of the corrosion oxide film which contained the bound-up radioactive species, even if this film was tightly adherent and uniformly distributed throughout the inside of the tubing. Utilizing this calculation an approved type B container would have to be employed for transport of the generator. Fabrication and certification of this container would have been impractical. Instead a one time exemption was applied for through the Department of Transportation to ship the generator via barge as LSA with the generator shell serving as the container. The exemption request was supported by a hazards analysis showing that an accident significant enough to breach the generator resulted in only a minor release. The radiation calculations demonstrating compliance with the rules in Title 49 of the Code of Federal Regulations and a detailed operation plan for loading, securing, transporting and unloading were also submitted. Approval by the D.O.T. staff was given without comment, and no substantive comments were received during the two public comment periods.

Because of the attention the project received due to intervenor actions, the Department of Energy (DOE) determined that a detailed environmental impact assessment would be required. The assessment covered all aspects of the program from acquisition thru disposal of the generator at the completion of the program. After an extremely thorough review the assessment was published and a negative declaration issued by the DOE office of NEPA affairs. The negative declaration meant that the environmental impact assessment was satisfactory and an enviromental impact statement would not be required.

All approvals were received by March 28, 1980 and work on the Surry site began April 14, 1980. The steam generator was removed from storage, the storage cradles removed and shipping cradles installed, and the generator and cradles were loaded onto a special trailer. The generator was transported from the storage facility overland about one mile to the Surry site shipping/receiving dock and loaded on to the barge April 17, 1980. The generator was secured to the barge deck by welding and a restraining system that would withstand simultaneously a 1.5g acceleration in any direction, 90 knot winds and a ten foot wall of water striking the generator broadside.

The Coast Guard made a preliminary survey on April 18, 1980 to assure compliance with the radiation levels. Regulations require less than 10 mr/hr, six feet from the edge of the barge. Per the calculations in the Operation Plan, the levels were less than one mr at the edge of the barge. The original intent had been to install lead shielding on the generator, however, an analysis showed that more exposure would be accrued installing lead shielding than from loading the generator on the barge with no shielding. An alternate shielding concept was evolved that resulted in partial concrete walls being erected on the barge prior to loading the generator. This concept was more cost effective, met transportation licensing requirements and resulted in lower worker exposure. Figure 1 shows the generator and shielding on the barge. The final inspection by the Coast Guard was completed by 1200 April 20, 1980, and the barge pulled away from the Surry dock at 1600 with the high tide. The shift from the river to the ocean going tug was achieved by 1800 and the journey down the James River begun. The 6,000 mile route for the generator was down the James, into Chesapeake Bay, into the Atlantic Ocean south through the windward passage, across the Caribbean, to the Panama Canal, south through the canal then north along the coast to the Columbia River.

The voyage to the Panama Canal took ten days and the vessel arrived in Cristobal Panama April 30, 1980. The canal transit was completed May 5, 1980. A recording triaxial accelerometer monitored motions input to the generator from the wave induced movement of the barge. No accelerations larger than 0.1g were observed from Surry, Virginia to Panama. The voyage north was equally smooth and the generator arrived at the Hanford site May 25, 1980. Off loading required three days and the generator was placed in interim storage May 28, 1980.

Other accomplishments during FY 1980 included design completion of the Steam Generator Examination Facility (SGEF). This facility will be used to house the Surry IIA steam generator for nondestructive and destructive examinations. Figure 2 is an artist's rendition of the SGEF.

Prior to transporting the generator an examination was conducted through three holes cut in the generator shell at different elevations. This examination and subsequent analysis of corrosion product samples established that the unit was not so service degraded as to preclude successful shipment, and that storage at Surry had not resulted in damage that would affect the validity of research objectives.

PLANNED RESEARCH ACTIVITIES

Introduction

During the last two decades a large amount of experience has been gained in the design and operation of nuclear steam generators. Thus, measures to prevent recurrence of some degradation mechanisms found in the Surry generator have been incorporated in many newer units. However, there are a large number of steam generators in world wide service that have close design similarities to our research unit. In addition, due to the long lead time on nuclear plant construction there are units just starting operation that have features similar to those that caused operational problems in the Surry generator. Finally almost all steam generators have certain features in common, such as they must be nondestructively and remotely inspectable to reliably determine integrity of the component. In addition, there is a high probability that during their service life many generators will be cleaned or decontaminated. There are benefits which will accrue to all three catagories mentioned above by the proposed Surry generator research. Major fields of research will include nondestructive examination and testing (NDE and NDT), statistics, health physics, and materials. Figure 3 provides a list of tasks and the proposed schedule for this program. The following paragraphs provide background on the tasks listed in Figure 3.

History

An initial program task will be to establish a documented history of the generator, in as much detail as possible. This history will include all available in-service inspection records and operating data. Operations aspects we will seek to determine would include: water chemistry (secondary side) during lay-up, start-up and operation; blowdown flow rate variations and carry-over variations.

In addition, plant operating history aspects such as sodium to phosphate ratios and the number of steam generator tube leaks will be catalogued. Finally, plant design features which might influence generator life will be evaluated. Establishing a comprehensive background will enable a correlation of experimental results achieved in this program to similar problems arising in other nuclear systems.

Nondestructive Testing, Inspection, and Evaluation

Nondestructive testing (NDT) is a vital ingredient to safe, efficient operation of nuclear reactors. In pressurized water reactors, one of the main NDT efforts is assuring that the primary system pressure boundary remains integral in the thousands of relatively thin (0.035-0.050") heat transfer tubes in the steam generators. In the past, this has involved use of single frequency eddy current

examinations at NRC mandated intervals of an NRC mandated number of tubes. However, there are recognized shortcomings in the method and methodology of inspection which are now being addressed. Tube defects in certain regions cannot be detected by conventional single frequency eddy current tests. Also certain types of defects have either limited detectability or a large degree of uncertainty when detected. In addition, there appears to be no published data of the potential effects of steam generator cleaning on NDT defect characterization.

To improve detectability of defects, several new generation eddy current techniques such as multifrequency, multiparameter units and new probe designs have been devised. These units need to be tested in a manner that will allow qualification of their use. In addition, NRC is evaluating needs for upgrading its steam generator ISI inspection criteria. To assure an integral primary system pressure boundary, what is a statistically valid sampling of tubes? In order to establish a statistical basis of examination important questions concerning NDT inspection measurement error and confidence limits must be established. Only then can a statistical evaluation of in-service inspections provide valid information on tube degradation rate and defect distribution.

The NDT tasks outlined below are designed to answer the questions raised. We propose to examine a large percentage of the generator tubes by as many NDT systems as can be obtained. A direct validation of NDT results will then be conducted by selective sectioning of generator tubes. This will provide comparison data between instruments and techniques. It will also establish a large validated statistical basis on which to evaluate NRC inspection requirements. It is our current intention to maintain highly characterized sections of the generator for future NDT development work and validation testing.

Another programmatic area utilizing nondestructive techniques will be a comprehensive examination of the secondary side of the generator. This effort will involve cutting selective holes in the generator shell to facilitate visual inspection and sampling. In addition we will utilize boroscopes, special minaturized cameras and fiber optics. The object is to establish as comprehensive a picture as possible of sludge/corrosion product distribution, condition of tube support plates, and location of defects which may or may not be detected by eddy current tests.

The identified NDE task areas are as follows:

1. Examine all available data tapes, analog records and service notebooks of past Virginia Electric Power Company (VEPCO) Surry steam generator inservice inspections, to help establish a statistical program plan.

2. Conduct a baseline eddy current examination of all unplugged tubes in the steam generator after placement in the Hanford research facility. This will determine possible changes in indications since the last ISI due to service or transport damage.

3. After unplugging selected tubes, conduct nondestructive examinations of the generator using all obtainable technologies. This will create a comprehensive bank of comparable NDE data for the various techniques. Included would be new generation eddy current techniques (multifrequency, adaptive learning, new probe designs) and other NDE techniques (ultrasonics, fiber optics, vibration devices, etc.).

4. Conduct a comprehensive optical examination of the secondary side of the steam generator.

5. Conduct pre and post cleaning experiment NDT to determine changes in generator tubing defect indications and the effects on NDT ability to characterize defects. This would initially involve laboratory studies on specimens removed from the generator.

6. Establish a small-scale tube bundle that will contain a collection of defects indicative of those found in the Surry generator. This mock-up will be used in qualifying future NDT equipment, inspection procedures and repair methods.

Statistics

Statistical interpretation of the large amounts of data to be provided by this program will form the basis for much of our scientific analysis. Therefore the earliest task in this area will be establishment of an information storage and retrieval system that will facilitate later analyses. Next a statistical analysis will be conducted at the front end of the program to determine a valid sample of various defects necessary for confirmatory evaluation of NDT techniques. This analysis, coupled with historical ISI data, will determine which plugged tubes in the generator should be unplugged to provide a good defect sample. Tubes that have their plugs removed will be examined by eddy current inspection to determine if the defect as characterized prior to plugging has changed. This input will serve to modify as necessary the statistical defect analysis for NDT validation studies.

As mentioned earlier, there is a need to confirm what is the requisite extent of inservice inspection (in terms of percentage and distribution of tubes) that will satisfy regulatory requirements for safe operation. An extension of this work could also help establish the inspection extent that would optimize reactor availability at a minimum radiation exposure for inspection. In order to accomplish these goals statistical evaluation of several aspects is required.

The measurement error and confidence limits of any nondestructive technique allowed for ISIs must be determined. Also there is need to know what constitutes a statistically valid sample of steam generator tubes to examine. A concentration of defects in certain tubes or tube locations would certainly influence the choice of how many tubes from certain areas would constitute a statistically valid sample. Only with a statistically valid model of generator defects, can a reliable analysis be made of the rate of degradation, the distribution of defects, and the probability for unscheduled outage due to tube failure. This model could also serve to improve and/or confirm the safety analysis associated with postulated reactor accidents. It is realized that a model based on defects in the Surry generator is not going to be universally applicable to all other nuclear steam generators. On the other hand, used wisely, selected aspects should be widely applicable.

Most of the statistical data will be obtained as an off-shoot of the various destructive and nondestructive examinations. The research program would provide data on:

1. Steam generator tube wastage pattern, extent and elevation;
2. Tube denting pattern, extent and crack pattern;
3. Tube support plate deformation and crack pattern;
4. Tube to tube sheet condition;
5. Sludge location and extent of pile up;
6. Corrosion product distribution;
7. Information on accuracy of various NDT techniques;
8. Statistical analysis of the above data in light of regulatory requirements for safe operation.

Materials Research

Materials research is divided into several categories: mechanical testing of defected tubing specimens to determine remaining strength; determination of effects of defects on reactor operation, e.g., leak rates; analysis of corrosion products and determination of corrosion mechanisms; and research/demonstration of the feasibility of and techniques for secondary side cleaning and primary side decontamination. Mechanical testing including burst and collapse testing under simulated operating and accident conditions will confirm the predictive models of tube failure developed in the NRC Steam Generator Tube Integrity Program. These models were based on tests conducted on mechanically and chemically simulated generator tubing defects. Service defected tubing is necessary to prove that appropriate defect

simulations were used in developing the predictive failure models. Also that defect combinations are not present in the generator that would result in tube failure at pressures below the lower bound predicted.

The association of leak rate data with various defect types and severity will provide guidance on allowable defect degradation for safe operation. Future NDT developments will be able to provide more information about defects. An ability to provide length and depth information and distinguish more clearly between defect types may lead to different plugging criteria for different defects. These criteria will have to be based on the consequences of tube failure with the various defects. Specimens will be removed from the generator that will allow separate leak rate determinations. These specimens will be either identified thru-wall defects, or defects that can be extended thru-wall. Of particular interest will be inner-row U-bends, potential cracking under dented areas, and leakage from the tube sheet crevice region.

Analysis of location, extent and type of corrosion products will answer many questions. Defects in tubing can be associated with specific corrosion processes. More knowledge about the processes and where they occur will lead to a better understanding of how to prevent further deterioration or eliminate the problem in new units. Detailed information of corrosion product location will also provide insight into the thermal hydraulic aspects of generator function. Hot spots and steam blanketing regions could be identified leading to design improvements.

Steam generator cleaning relates directly to the existence of corrosion products. If the corrosion products could be successfully removed, there is the possibility of slowing or stopping further generator deterioration. For example, removal of the magnetite deposit in the annulus between steam generator tubes and the tube support plates would alleviate the denting problem and cracking of the support plates. Likewise removal of sludge deposits will affect tube thinning, wastage and stress corrosion cracking mechanisms. There is also the probability that generator decontamination will be pursued to reduce radiation exposures during maintenance and repair of generators. The cleaning and decontamination processes must not seriously affect future serviceability of components. Thus, there are several aspects to be addressed in cleaning/decontamination research. These include testing the effectiveness of proposed methods in a real environment, assessing the effects on serviceability of a unit exposed to cleaning/decontamination schemes, and regulatory issues. NRC will be expected to provide regulatory direction on acceptable decontamination and cleaning procedures. As input to formulating the regulatory guides, information must be developed on the effect of various cleaning procedures on future serviceability and safety. It is our intent to maintain a portion of the generator as an integral unit to study primary side decontamination and secondary side cleaning. A sufficiently large section will be isolated to create a realistic service environment for cleaning/decontamination studies.

The following is a summary list of materials tasks designed to answer the above issues and provide support to NDE and statistics requirements.

1. Burst and collapse strengths will be determined for specimens having a range of service induced wastage and/or cracks.

2. Leak rate, primary side conditions to ambient air, will be determined on tubes with inservice through-wall or extended to through-wall defects. This will allow a correlation of typical leak rates to type and extent of defect.

3. Tubes will be metallographically sectioned to provide information on service induced thinning, cracks, and cracks possibly masked by the tube support plates and support plate corrosion products. This task will serve to validate accuracy and reliability of previous NDE tasks. It will also provide information on possible effects associated with cleaning corrosion products from the annulus between the steam generator tubes and the support plates.

4. Examine U-bend sections to provide strain characterization. This information will show the extent of plastic deformation in different U-bend regions. The strain information will be valuable to determine areas subject to eventual failure, hence requiring closer monitoring. Correlation of the presence of inner diameter nucleated cracking with U-bend strain will be attempted. Information on the uniformity of strain characterization between numerous similar samples will establish if it is possible to generalize findings about U-bend condition in a unit from removal of a few specimens.

5. A statistical cross section of tubes will be destructively examined to provide data to correlate with eddy current information. Examination will be by sectioning and metallography.

6. Remove a section of the tube sheet for detailed destructive assay of the crevice region between tube and tube sheet. Correlate the destructive assay with prior NDT work and leak rate experiments.

7. Analyze sludge profiles and chemical species present at different levels. This analysis will assist development of suitable cleaning solutions. It should also provide information of how operating chemistries effect sludge generation and will allow development of water chemistry control guides. Correlations between sludge profiles and areas of corrosive attack on tubing will be established.

8. Chemical cleaning development/demonstration of removal of magnetite corrosion product between tubes and tube support plates. Alternately demonstrate a process that will alleviate or stop further denting, but may not remove the annulus corrosion product. Laboratory experiments on removed tube sections will provide validation of proposed cleaning solutions; assess necessary exposures and agitation; determine metal losses.

9. Laboratory studies employing specimens removed from the generator on primary side decontamination. Utilize chemistries and methodologies developed in the program and/or those proposed by other contractors to assess decontamination factor, evaluate metal losses and determine effects on the structural integrity of stressed/strained tubes. These studies will also examine decontamination of the channel head.

10. Reconstruct thermal-hydraulic characteristics relevant to degradation patterns, e.g., steam blanketing regions compared to areas of chemical concentration and generator flow profiles.

11. Chemically clean the secondary side of an isolated portion of the steam generator. Utilizing solutions evaluated in '8' above perform a cleaning operation with dilute solutions. Optically analyze effectiveness of cleaning operation. Attempt to improve any necessary techniques and equipment. Perform further cleaning task operations if initial effort achieves less that a predetermined desired result. Final cleaning operations will involve strong solutions to attempt maximum removal of radioactive species. Between cleaning operations perform eddy current evaluations on steam generator tubing and optically determine effect on tube support plates and other secondary side features. Experiments will also be conducted to assess the short term corrosion effects and long term potential for crevice corrosion and stress corrosion cracking due to the cleaning process.

12. Vibration testing following chemical cleaning. Analyze potential operational effects, such as vibration induced fatigue cracking, due to removal of annular deposits in support plates.

13. Decontaminate a portion of the steam generator primary side using solutions developed/evaluated in '9' above. Determine the decontamination factor and metallurgical effects of the decontamination.

14. Search for new modes of failure. Conduct a detailed general examination of several tubes.

15. Materials recovery. Large amounts of expensive and increasingly scarce resources are committed to nuclear components. Reuse of these materials for further nuclear applications or possibly non-nuclear commerce is desirable. We propose to seek sponsorship for efforts to extend current technology for the decontamination and recovery of zirconium to the many tons of Inconel 600 in the steam generator.

Health Physics

A necessary part of our research will be interaction with health physics personnel. This interaction is expected to develop into health physics oriented research. Specific areas of interest are in the determination of the distribution, and concentration of radioactive material in various parts of the steam generator. Of particular interest is the effectiveness of decontamination techniques in reducing the radiation field and in removing specific radionuclides. Sampling and measuring plans will be established and utilized to determine the distribution of radioactive material on the tubesheet and channel head, along the length of the tubes and between the hot leg and the cold leg. Other research would center on radiological aspects of cleaning, and decontamination, along with the waste and waste disposal problems generated by the different processes. Health physics would be closely involved in procedure preparation of many of the experimental tasks. This involvement will be documented to provide information that could be of use to others involved in steam generator maintenance, repair or cleaning operations.

LICENSING AND SAFETY ISSUES

NRC is responsible for establishing inservice inspection criteria for steam generators. Parameters such as intervals between NDT examinations, the number of tubes examined, criteria for plugging a defected tube, and to some extent instrumentation used to conduct inservice inspections are matters of NRC jurisdiction. The basic issue is primary system integrity and public safety, especially in the event of postulated accident scenarios.

The Steam Generator Program will:

- Provide verification of the accuracy and confidence bands associated with various NDT techniques used for inservice inspection.

- Establish the remaining integrity of service degraded tubing to provide input on tube plugging criteria.

- Establish whether defects exist in the generator that may not currently be detected by standard ISI practice. Evaluate the potential hazard of such a defect in terms of operational safety under reactor upset conditions.

- Provide input on the extent and distribution of tube degradation. This will aid NRC re-evaluation of regulations concerning the number of tubes inspected at a scheduled ISI.

- Conduct the first extensive evaluation of secondary side generator condition. Evaluate potential risks to generator integrity of secondary side failures. Provide input to potential regulations on secondary side inspections.

- Perform leak rate tests on service defected tubing. This will verify models used to establish licensing actions on permissible leakage during service, relate leakage to defect type, provide input to tube plugging criteria, and validate NRC accident analyses.

NRC has licensing purview over activities that could affect reactor system integrity or safety. It is anticipated that decontamination and/or cleaning of steam generators will be proposed for radiation reduction and corrosion inhibition attempts. NRC must evaluate units subject to decontamination/cleaning and license their return to service.

The Steam Generator Program will:

- Provide licensing input on the effects on long term system integrity of cleaning operations.

- Provide licensing input on the effects on long term system integrity of decontamination operations.

- Evaluate the health physics and radioactive waste aspects associated with cleaning, decontamination and decommissioning operations.

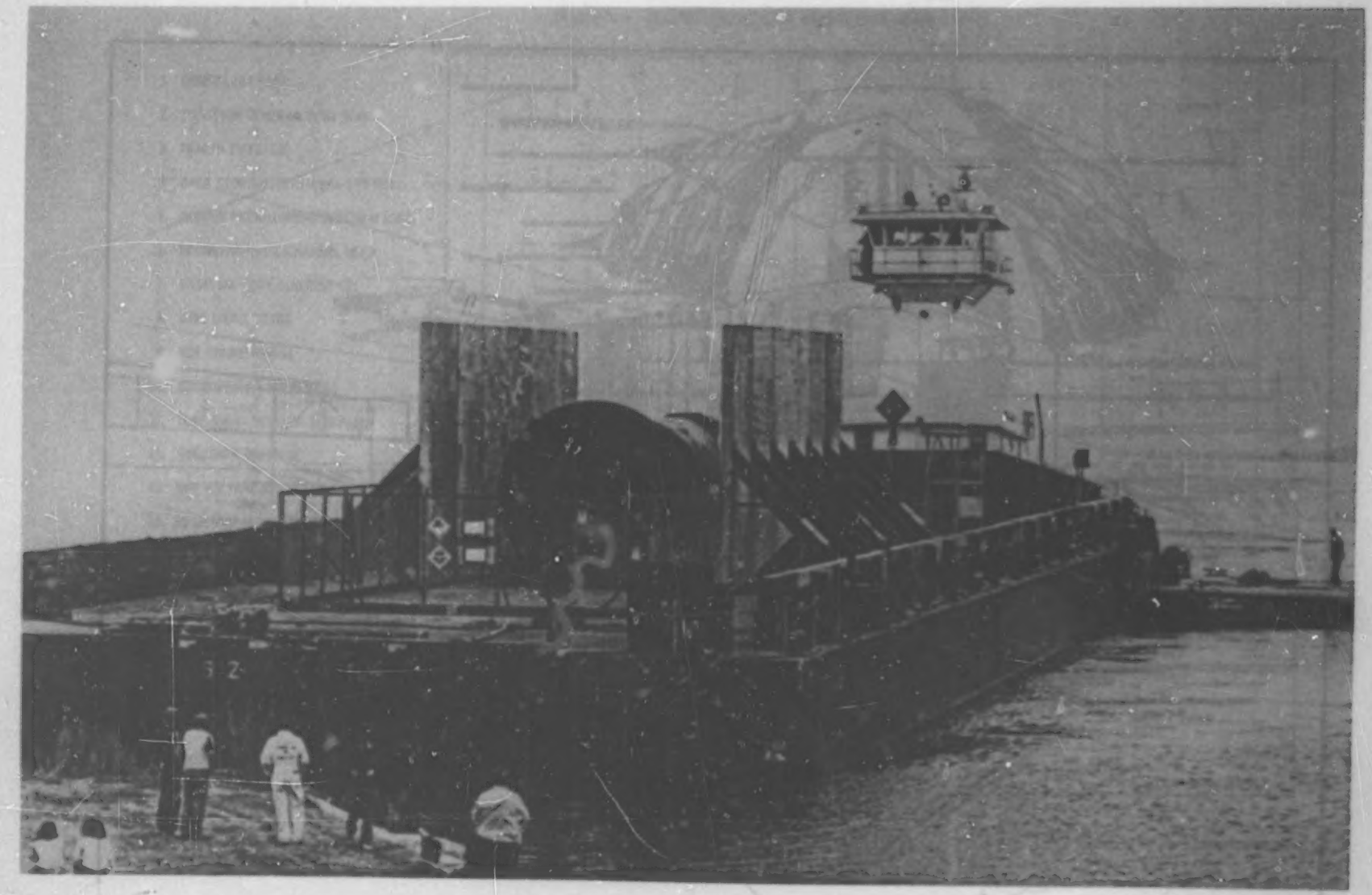


FIGURE 1. Steam Generator and Shielding on Barge

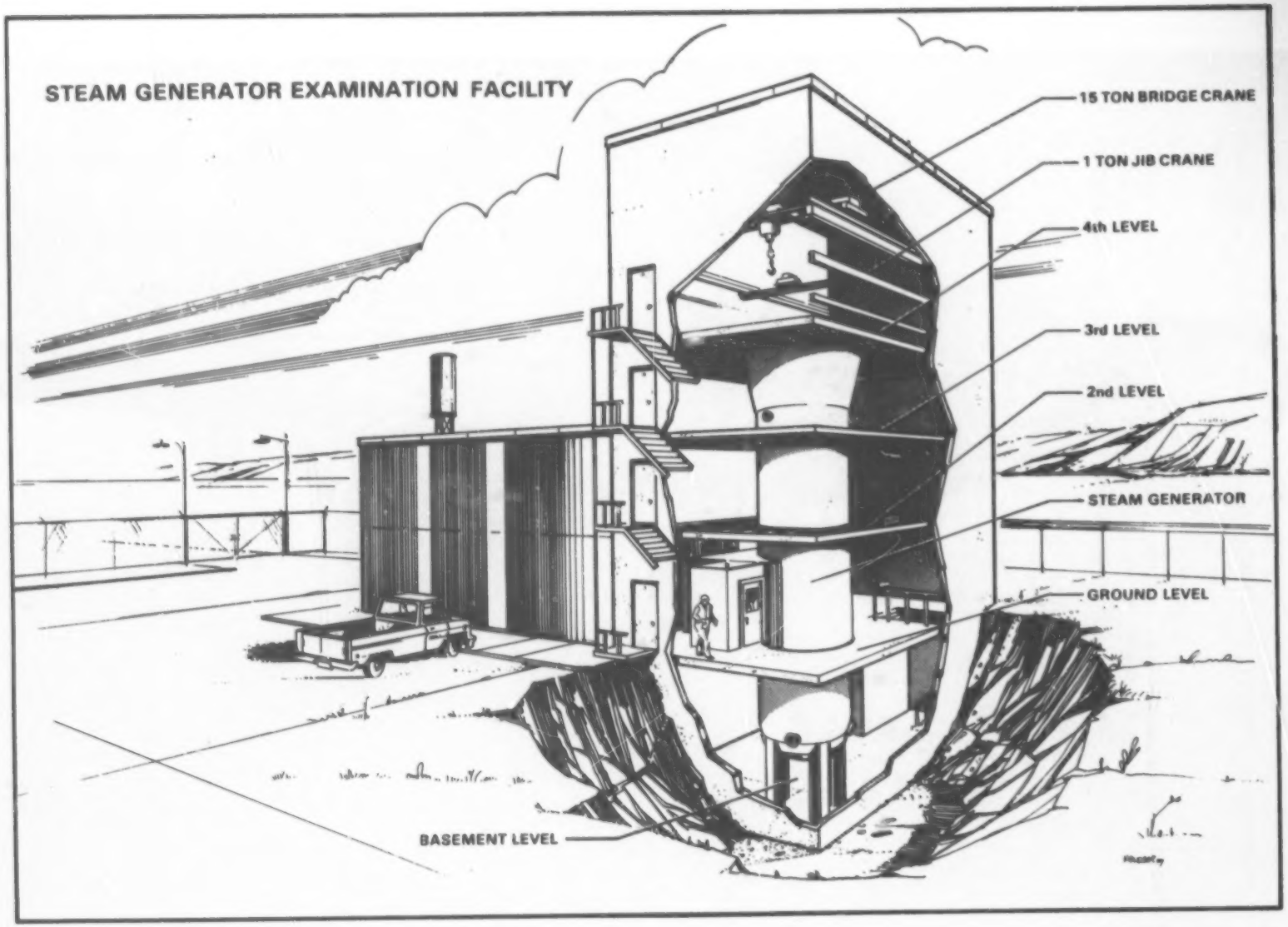


FIGURE 2. Steam Generator Examination Facility

pages tightly bound

MILESTONE CHART
SURRY GENERATOR PROGRAM

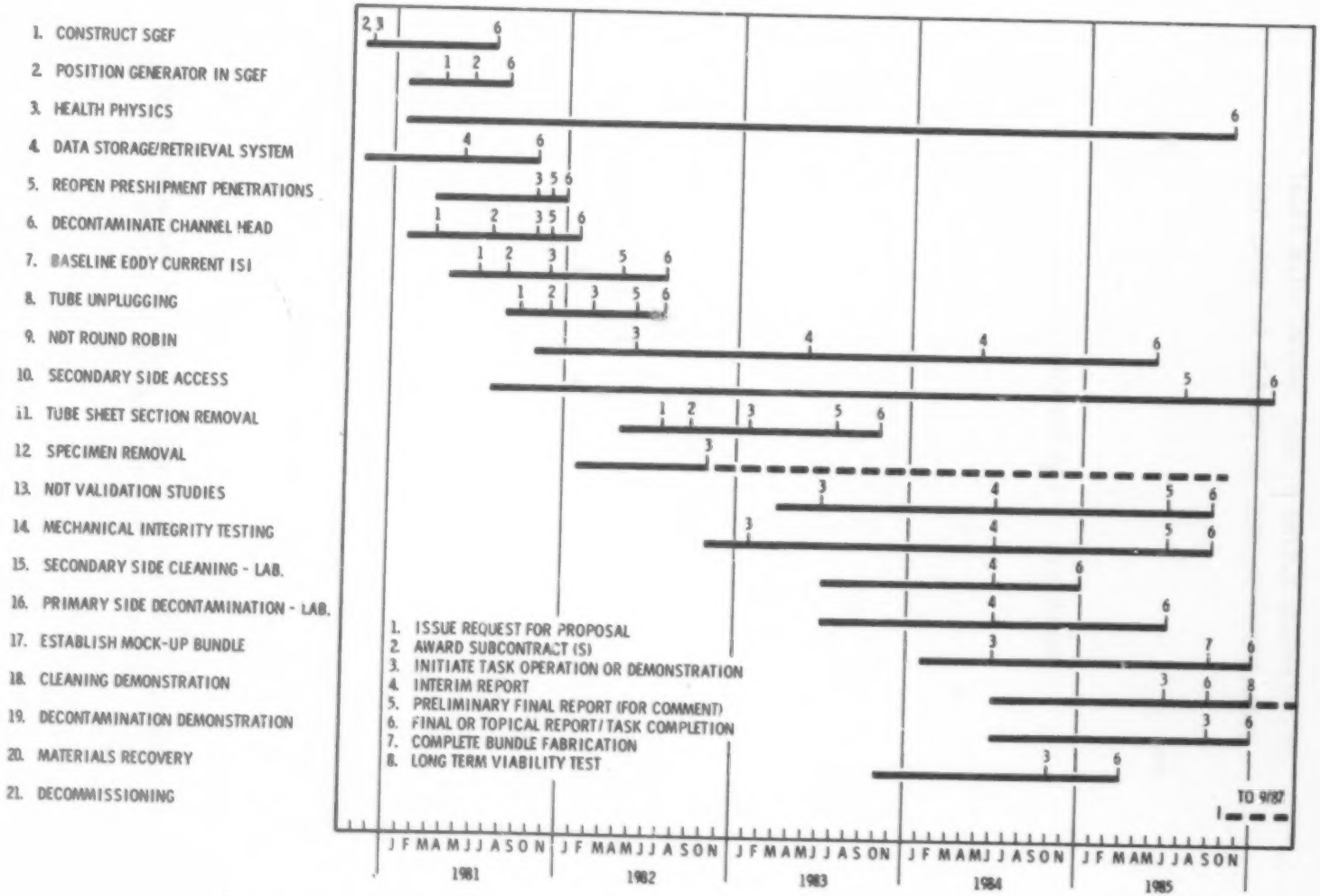


FIGURE 3. Milestone Chart Surry Generator Program

READEX FICHE best copy available

pages tightly bound

TESTS WITH INCONEL 600 TO OBTAIN QUANTITATIVE SCC DATA FOR EVALUATING SERVICE PERFORMANCE

T.S. Bulischeck and D. van Rooyen

October 1980

Corrosion Science Group
Department of Nuclear Energy, Brookhaven National Laboratory
Upton, New York 11973

Work supported by the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research

EXTENDED ABSTRACT

While it was shown over ten years ago in France that Inconel 600 can crack in high temperature water, performance of steam generator tubes has been remarkably good in the absence of abnormal conditions. Factors such as denting or abnormal chemistry of the secondary water, unfortunately have caused several stress corrosion cracks in steam generator tubes made out of Inconel and, since some of them caused primary to secondary leaks, the NRC has faced difficult licensing decisions because of the lack of quantitative information relating cracking to the various factors that contribute to SCC.

Because of the obvious need for a better basis for licensing in this regard, support for research work was provided by the Research branch of the NRC, and a program is under way at BNL to develop quantitative data on the stress corrosion cracking of Inconel 600 steam generator tubing. The objective is to develop a predictive model for estimating the life expectancy of steam generator tubing under normal or abnormal conditions. Variables under test include stress, strain, strain rate, environment, temperature, micro-structure and processing of the metal. The initiation and propagation stages of intergranular SCC are treated separately, and it is intended to generate the data so that they can be applied either to situations where deformation is active or where, for instance, some denting occurs but is arrested. The environments include pure H_2O, AVT, and primary conditions, as well as contaminants.

First tests were with tube type U-bend specimens in pure deaerated water at various temperatures. Intergranular SCC occurred in several, but not all heats of nuclear production grade tubing, with the shortest failure in less than two weeks at the highest test temperature, (365°C). The data collected thus far indicate an Arrhenius relationship between failure time and temperature. Tentatively, lower carbon contents have given lower activation energies in pure water for crack initiation, but tests in AVT and primary water are still to confirm this trend. Some U-bends tested in AVT have cracked in less than the time required in pure water.

Slow strain rate tests provide an accelerated test, by reducing the time needed for crack initiation. It causes continuous film rupture and also more closely duplicates the slow strain rates experienced by tubes that are being continually deformed by denting. Failures have occurred at all test temperatures (365°C, 345°C, 325°C, and 290°C) in pure deaerated water when strain rates of 10^{-6} sec^{-1} to 10^{-8} sec^{-1} were applied. An Arrhenius semi-log straight

line relationship exists for the first two heats tested, and the curves have the same slope, with an activation energy of about 39,000 cal/mole. A small amount of cold work was introduced in making some of the specimens and this accelerated both crack initiation and propagation. However, the cold working is not necessary to produce cracking, since differently shaped specimens (dual-ligament) also cracked, with the same temperature relationship but in times about twice as long. Crack initiation in the CERT tests is much different from U-bends, and gives an activation energy of about 7,000 cal/mole.

Other tests determined the effects of stress level and cyclic stress, but sufficient data are not yet available to allow an exact definition of the constants of the model that is being developed. Tentatively it seems that failure time is proportional to about the -3 power of applied stress in excess of the yield point.

Capsule tests have been operating to provide some insight on the effects of stress patterns. These capsules contain tight-fitting carbon steel plugs on which rapid magnetite growth is induced to deform the capsule walls. The outer walls of the capsules are exposed to primary water at 325°C and observations are being made for cracking. Other tests are being planned with whole pieces of tubing in order to check the validity of data from the work described above.

NRC LICENSING INTERESTS

ELIMINATION OF PRIMARY→SECONDARY LEAKS

PREVENTIVE PLUGGING OF TUBES:

DEFECTS: DEPTH IN TUBE WALL

DEFORMATION OF TUBES AT TSP

OVALIZATION AT U-BENDS

CAUSES OF SCC IN SERVICE

ABNORMAL SECONDARY WATER CONDITIONS

DEFORMATION: PRIMARY SIDE SCC

FATIGUE

BNL PROGRAM OBJECTIVES

SUPPORT FOR NRC NEEDS

DETERMINE QUANTITATIVE SCC DATA:

- METAL CONDITION
- ENVIRONMENT
- STRESS, STRAIN, STRAIN RATE

ENSURE VALIDITY OF MODEL

VARIABLES WHICH MAY INFLUENCE THE INCUBATION PERIOD OR PROPAGATION RATES OF INCONEL 600 SCC.

1. METALLURGICAL CONDITION

   A. HEAT TREATMENT (M.A.) (S) (T.T.)
   B. CARBON LEVEL (.01% - .05%)
   C. COLD WORK

2. CHEMICAL ENVIRONMENT

   A. PURE WATER
   B. PRIMARY WATER
   C. SECONDARY WATER (AVT)

3. TEMPERATURE

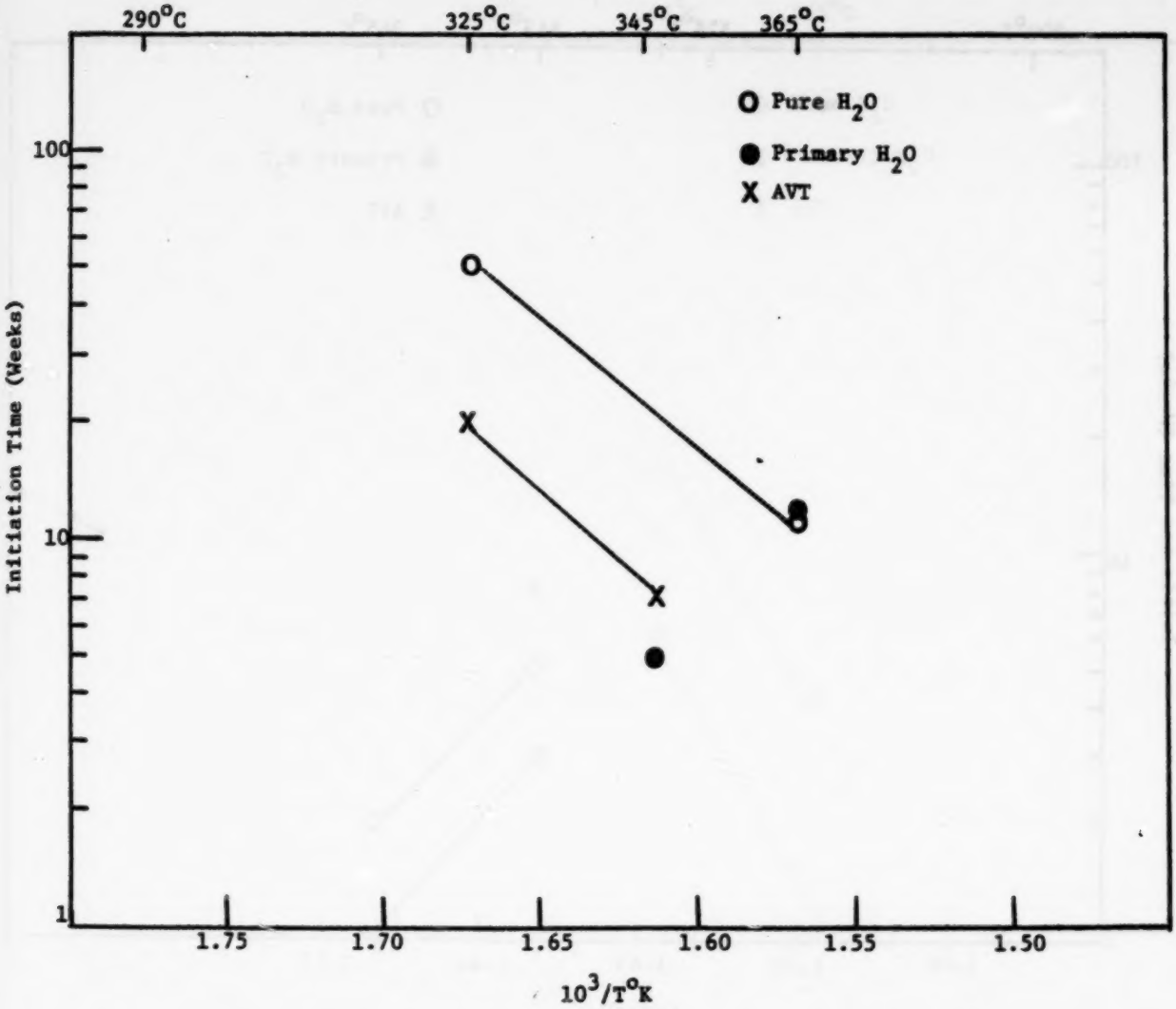
4. STRESS STATE

   A. STATIC

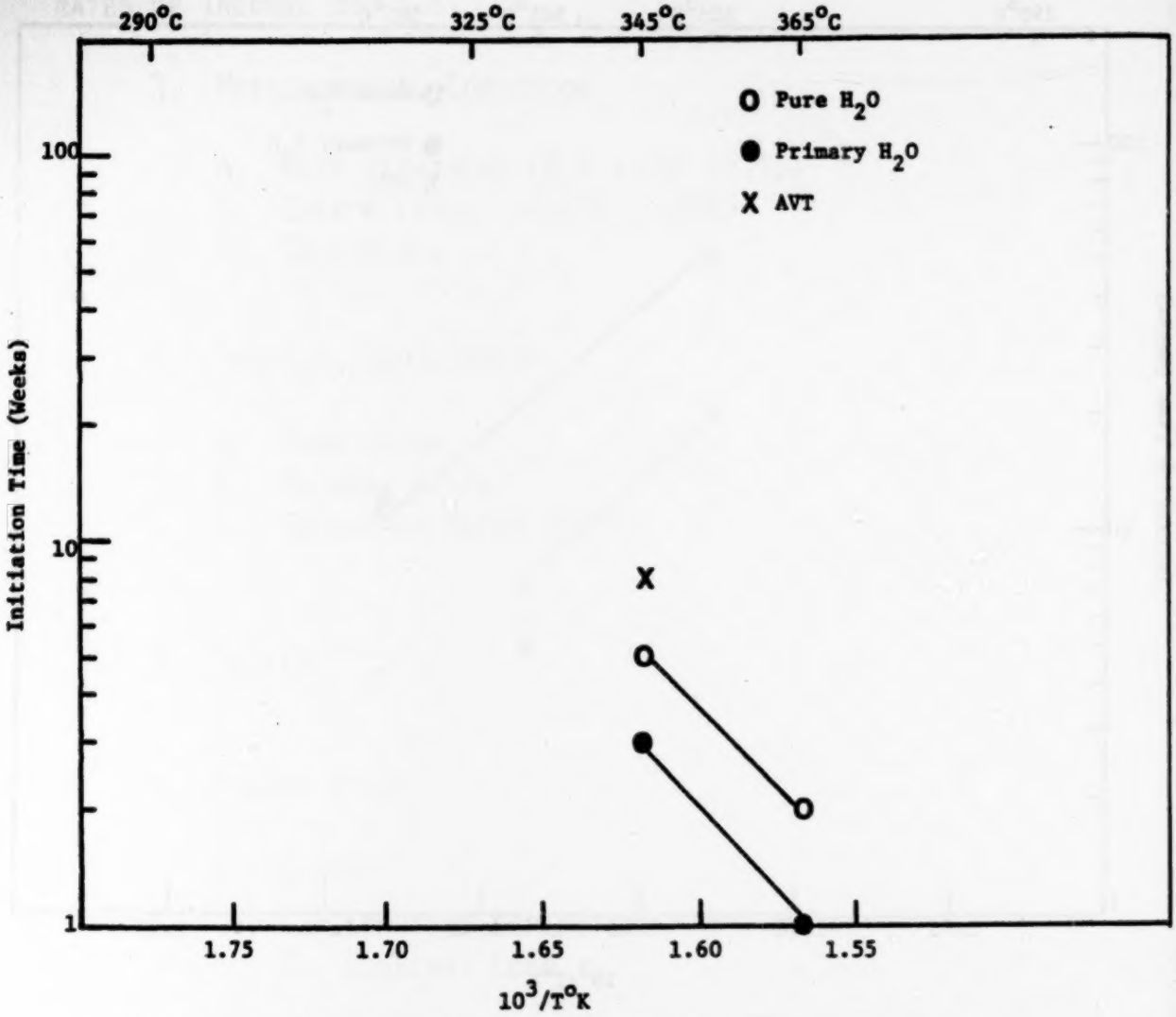
      1. CONSTANT DEFLECTION
      2. CONSTANT LOAD

   B. DYNAMIC

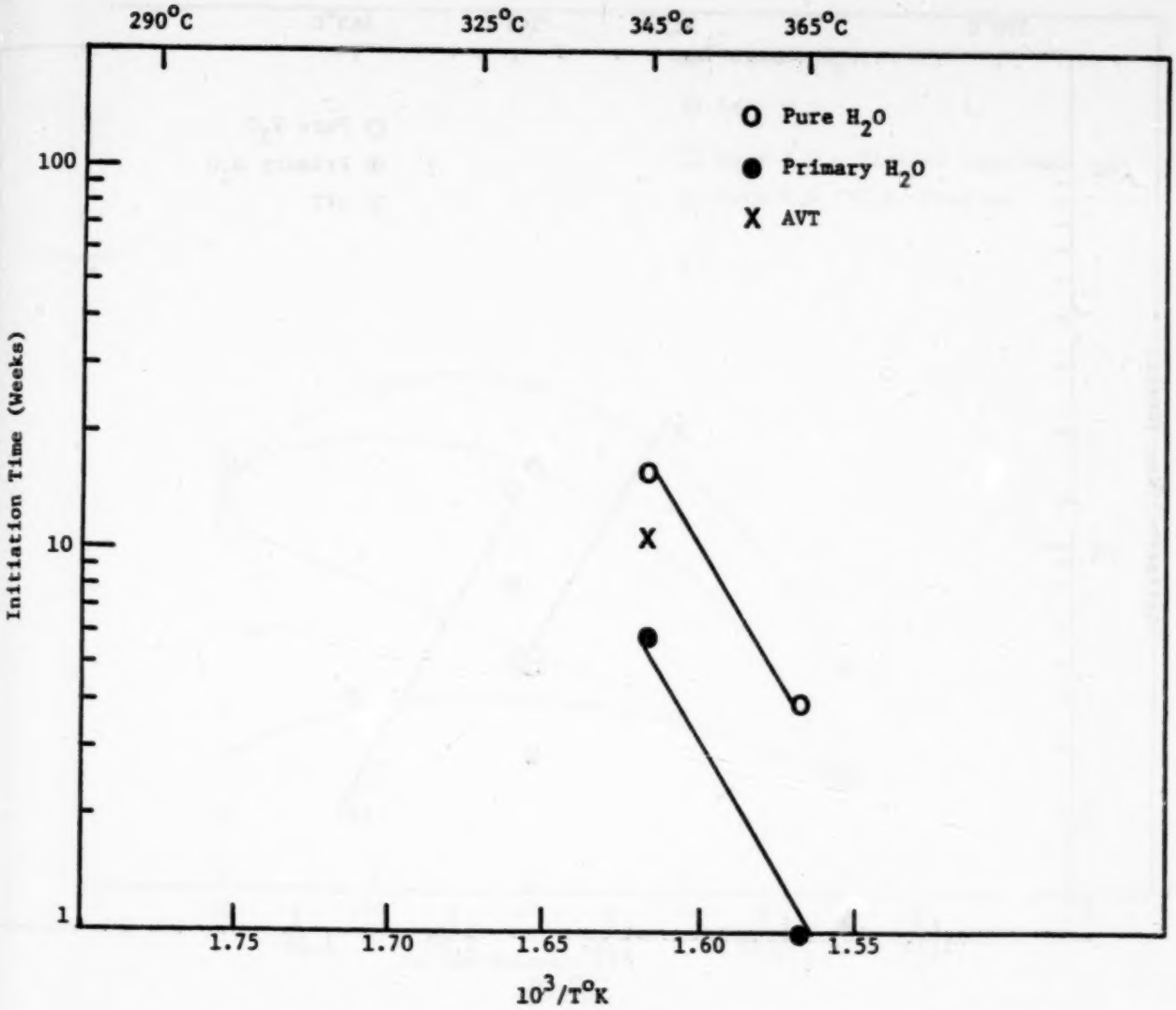
      1. SLOW STRAIN RATES
      2. CYCLIC LOADING



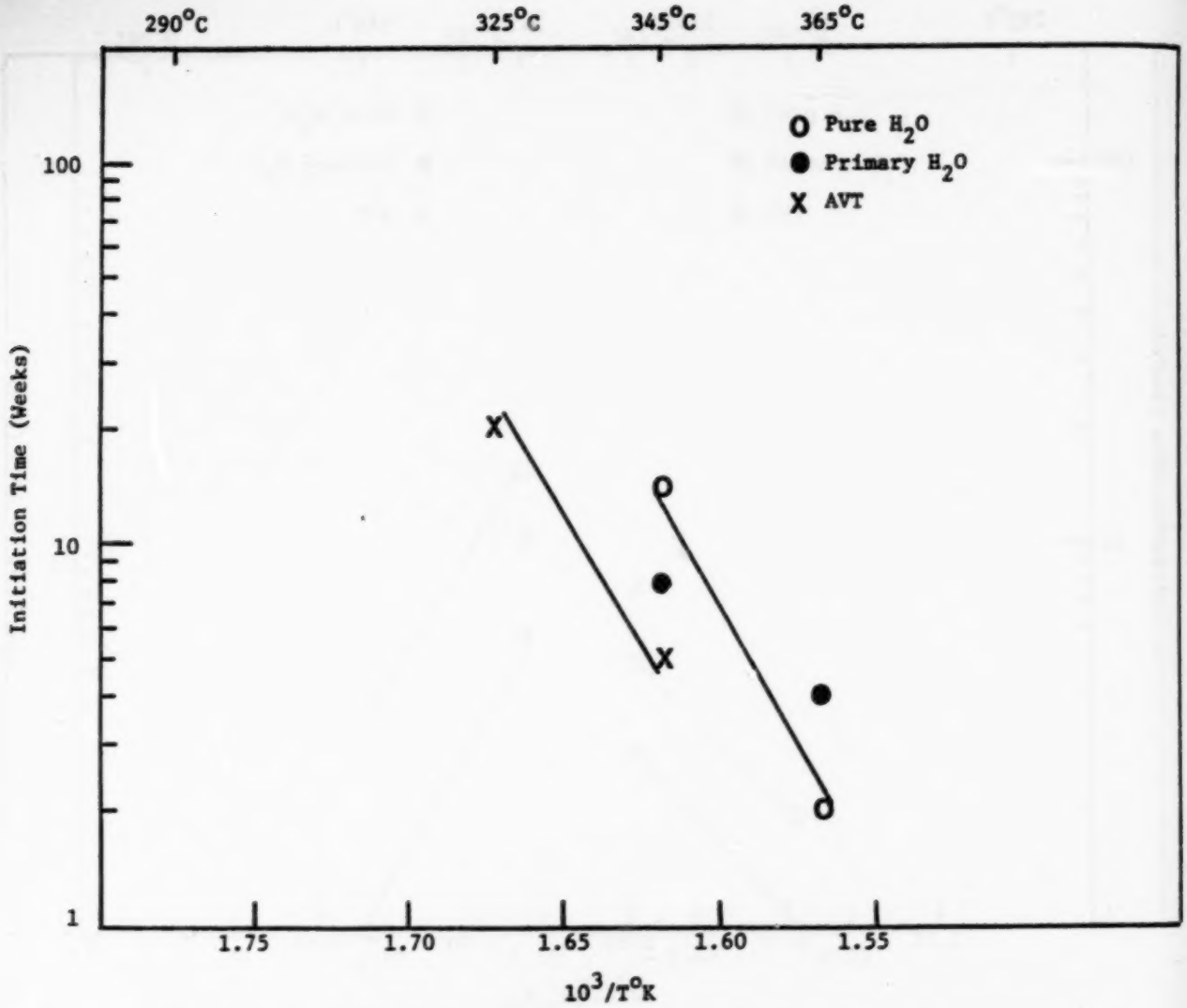
Effect of Temperature on Crack Initiation Time of .01% Carbon Material



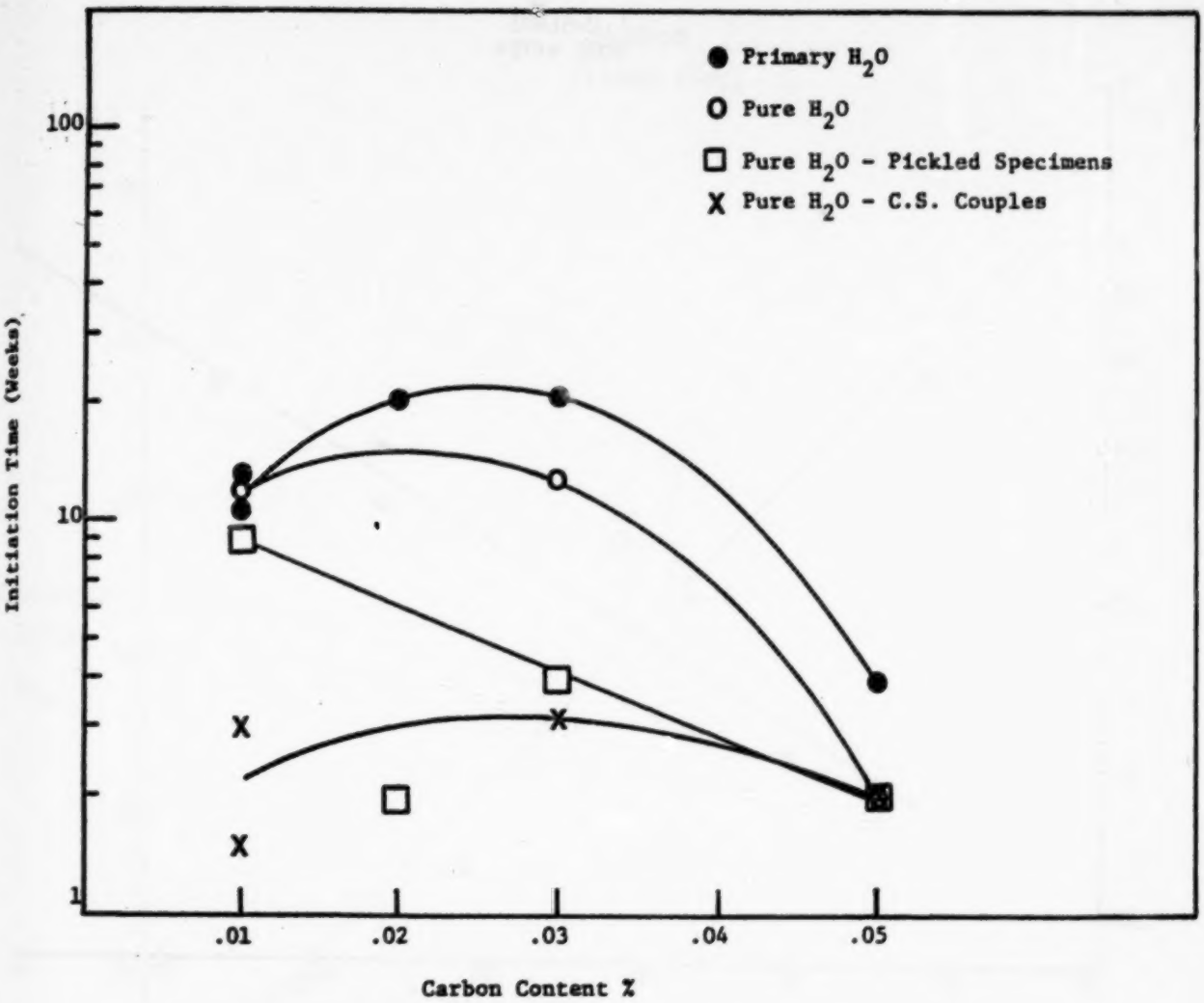
Effect of Temperature on Crack Initiation Time of .02% Carbon Material



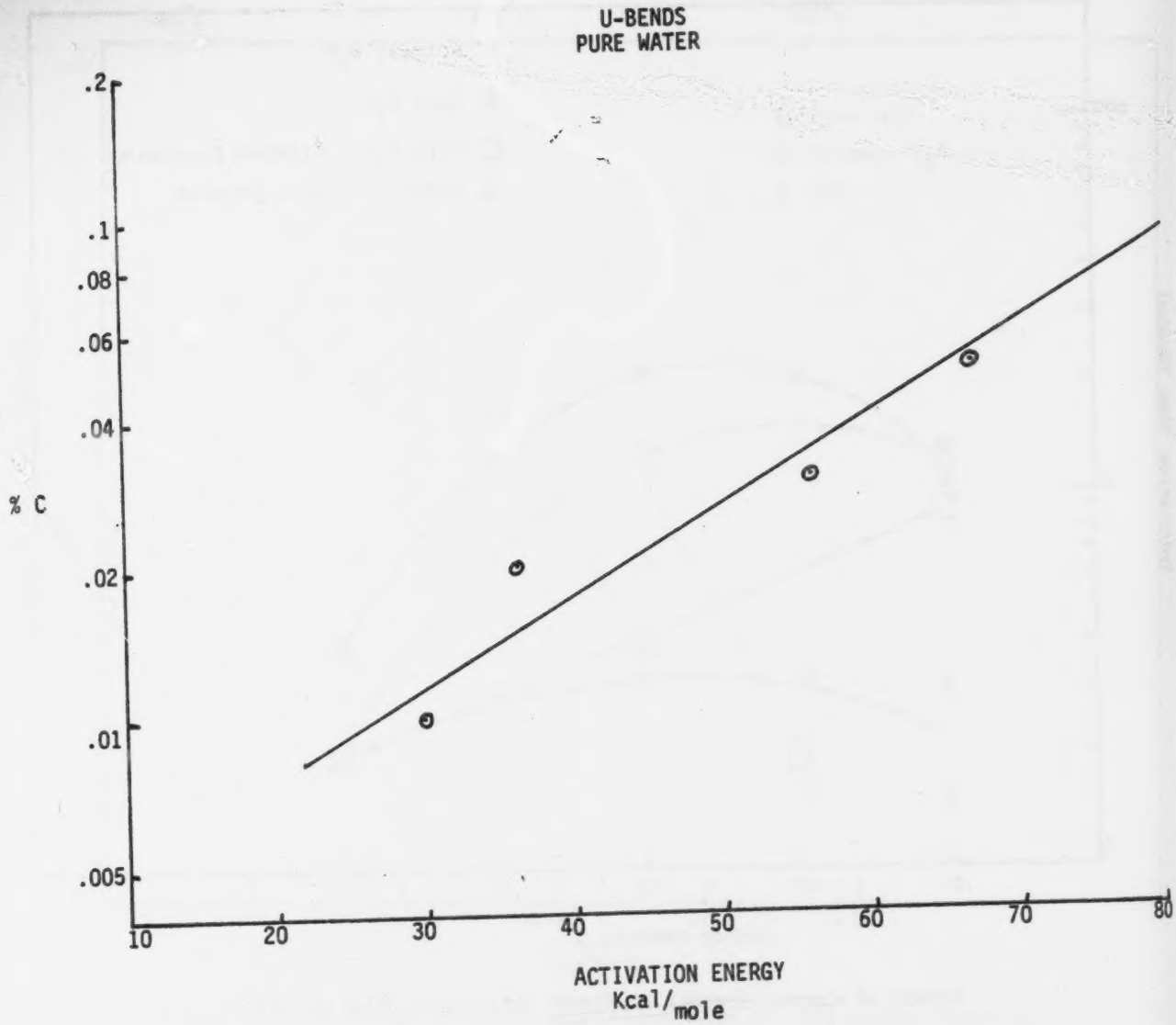
Effect of Temperature on Crack Initiation Time of .03% Carbon Material



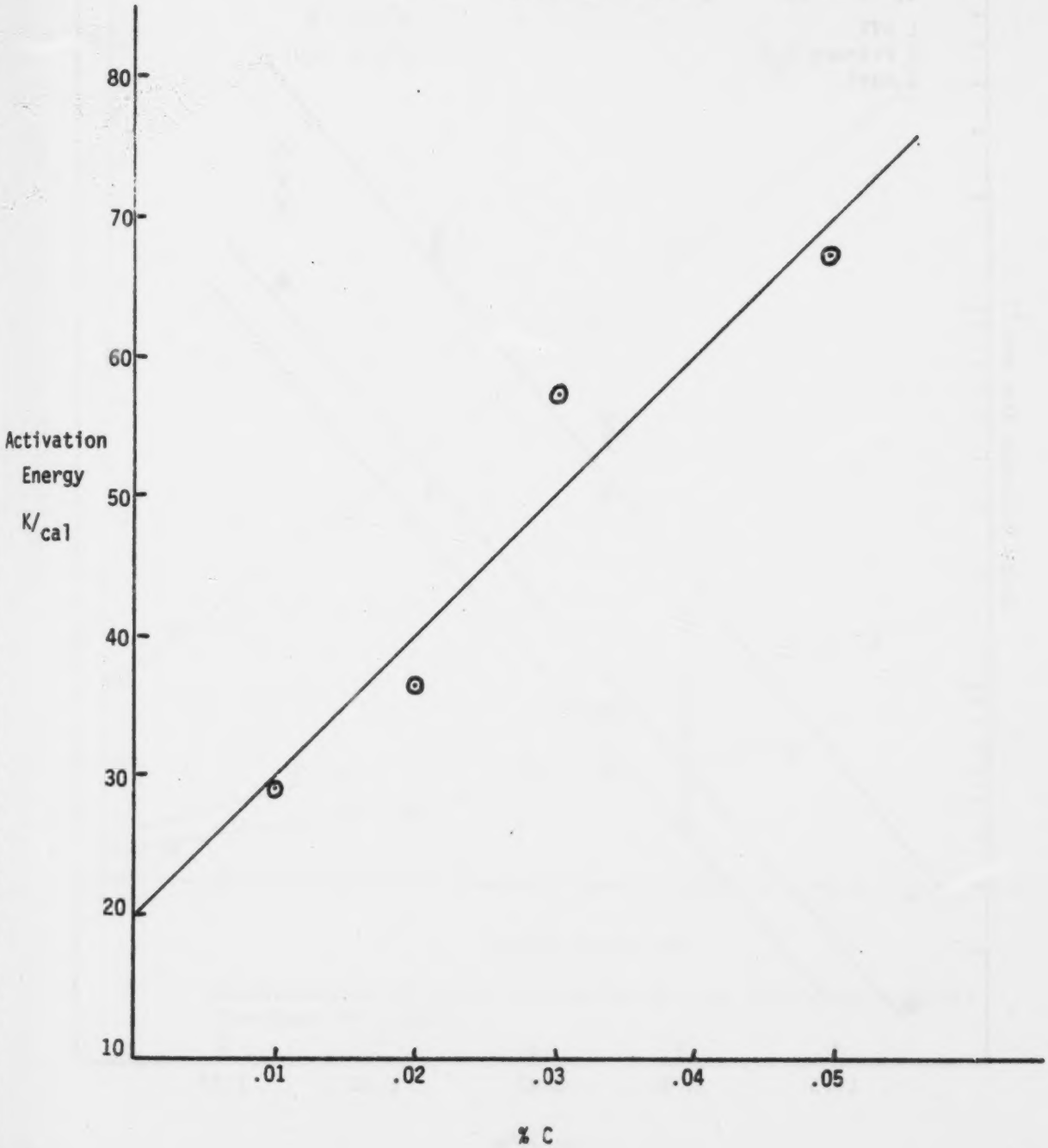
Effect of Temperature on Crack Initiation Time of .05% Carbon Material

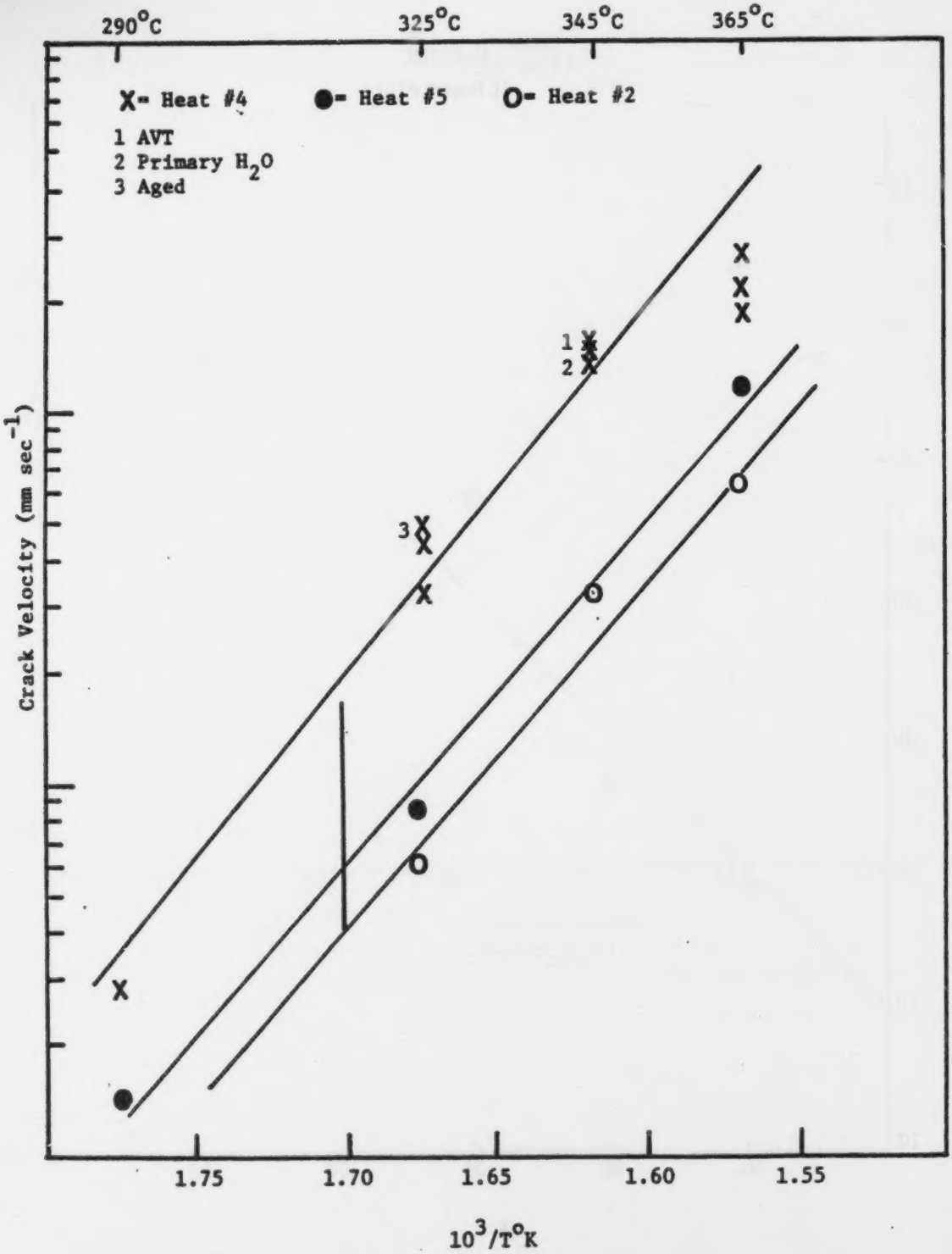


Effect of Carbon Content on Crack Initiation Time at 365°C

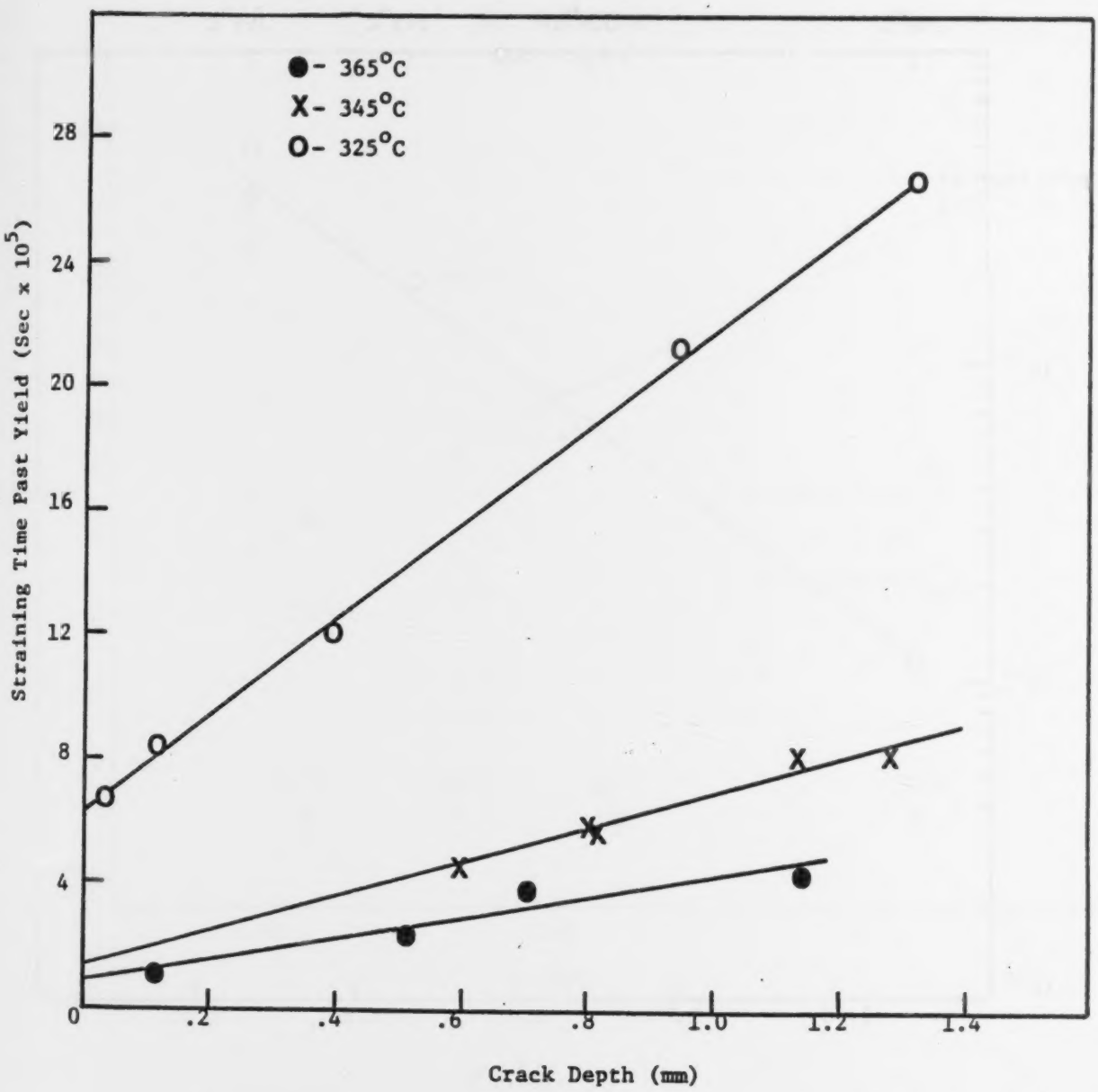

U-BENDS
PURE WATER
.2
.1
.08
.06
.04
% C
.02
.01
.005
10
20
30
40
50
60
70
80
ACTIVATION ENERGY
Kcal/mole

U-BENDS
(Linear Plot)

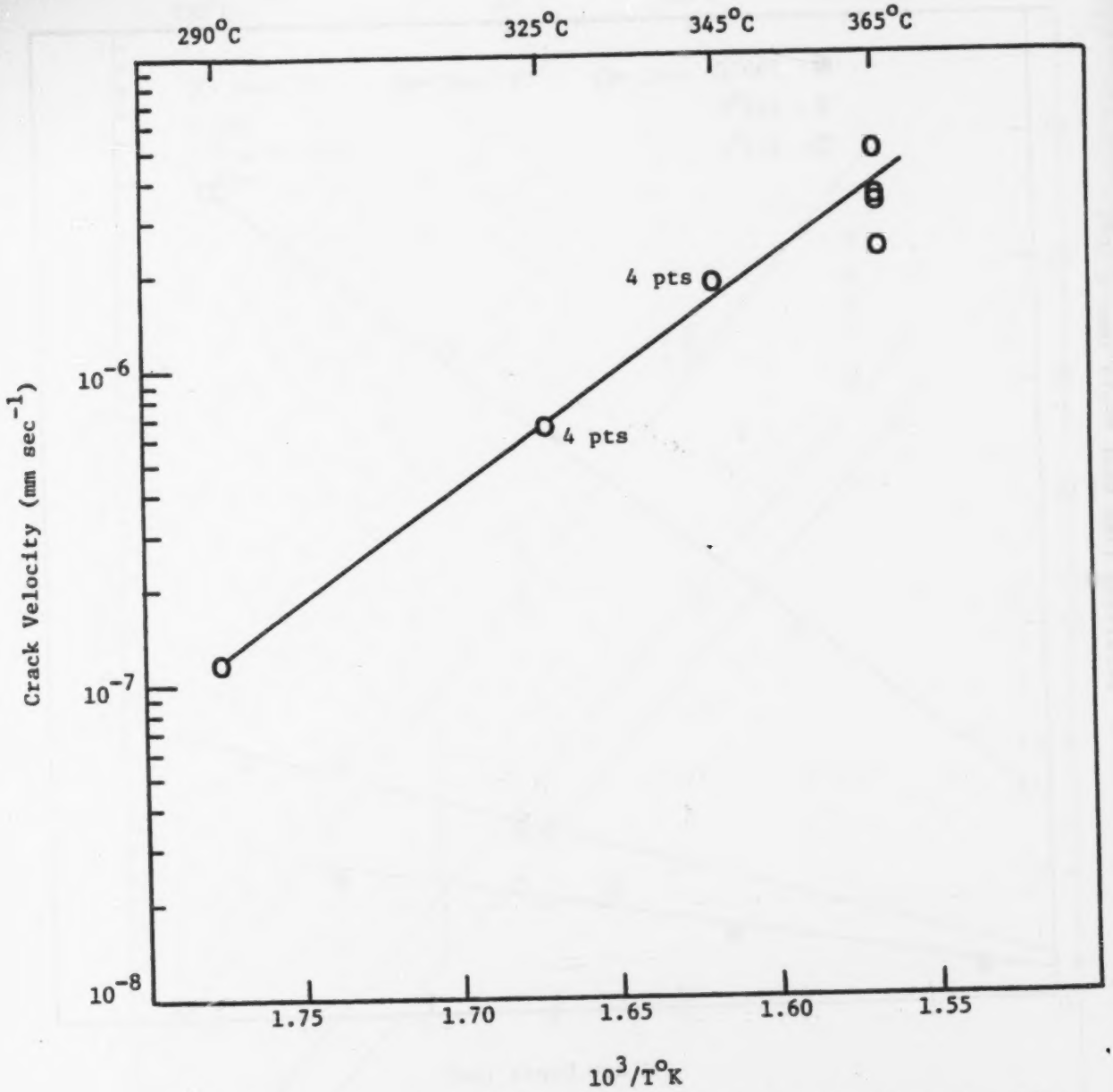




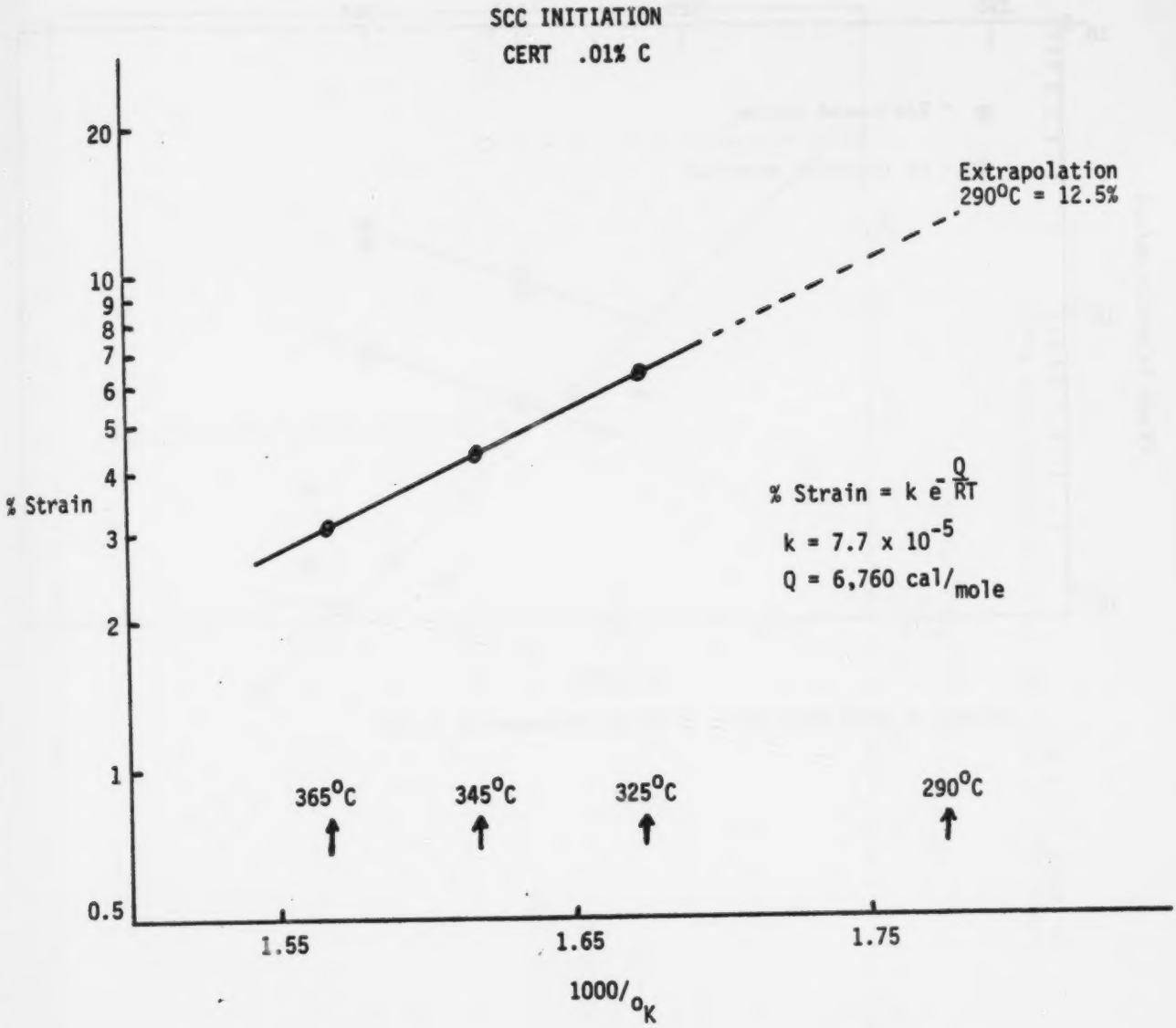
Effect of Temperature on Crack Velocities Determined Using CERT

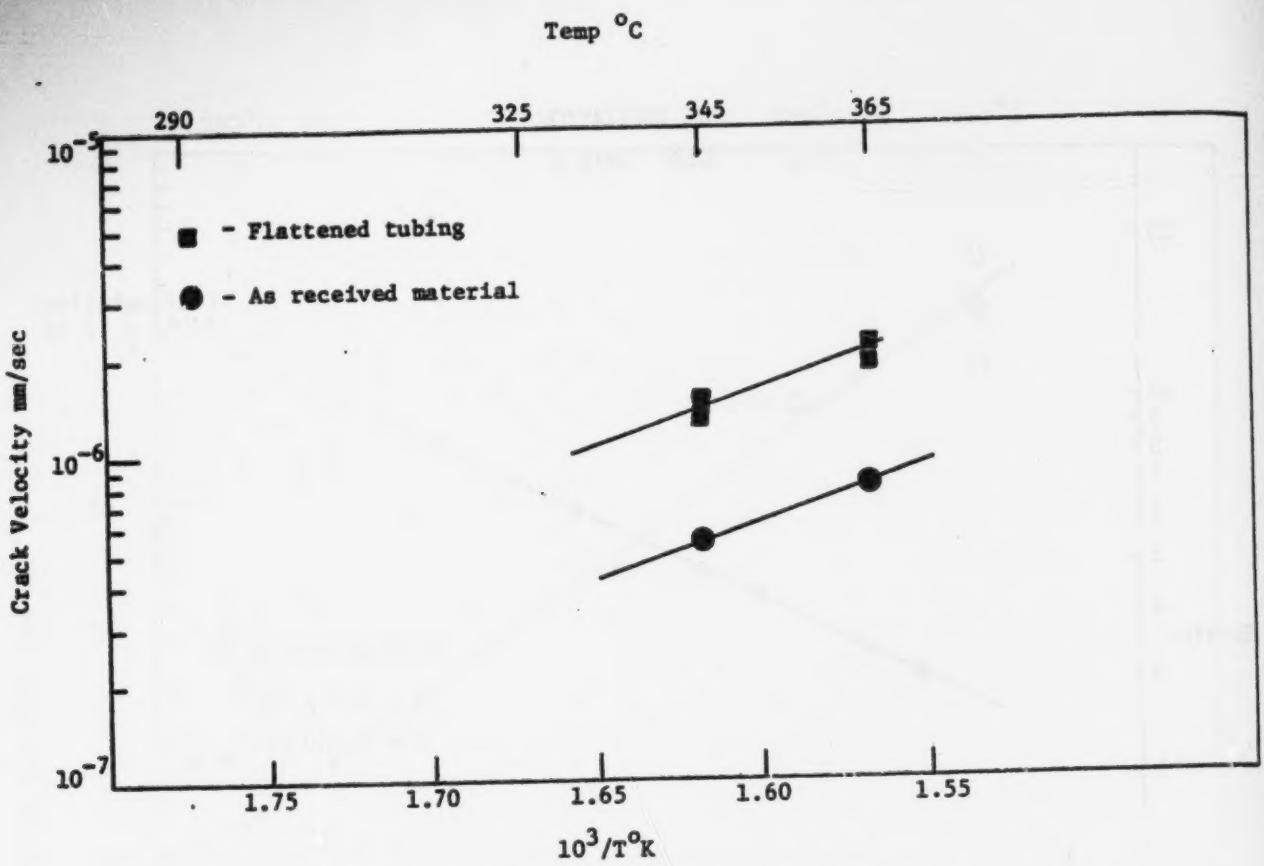


Determination of Crack Initiation in CERT Specimens Machined from Heat #4 (.01%C)

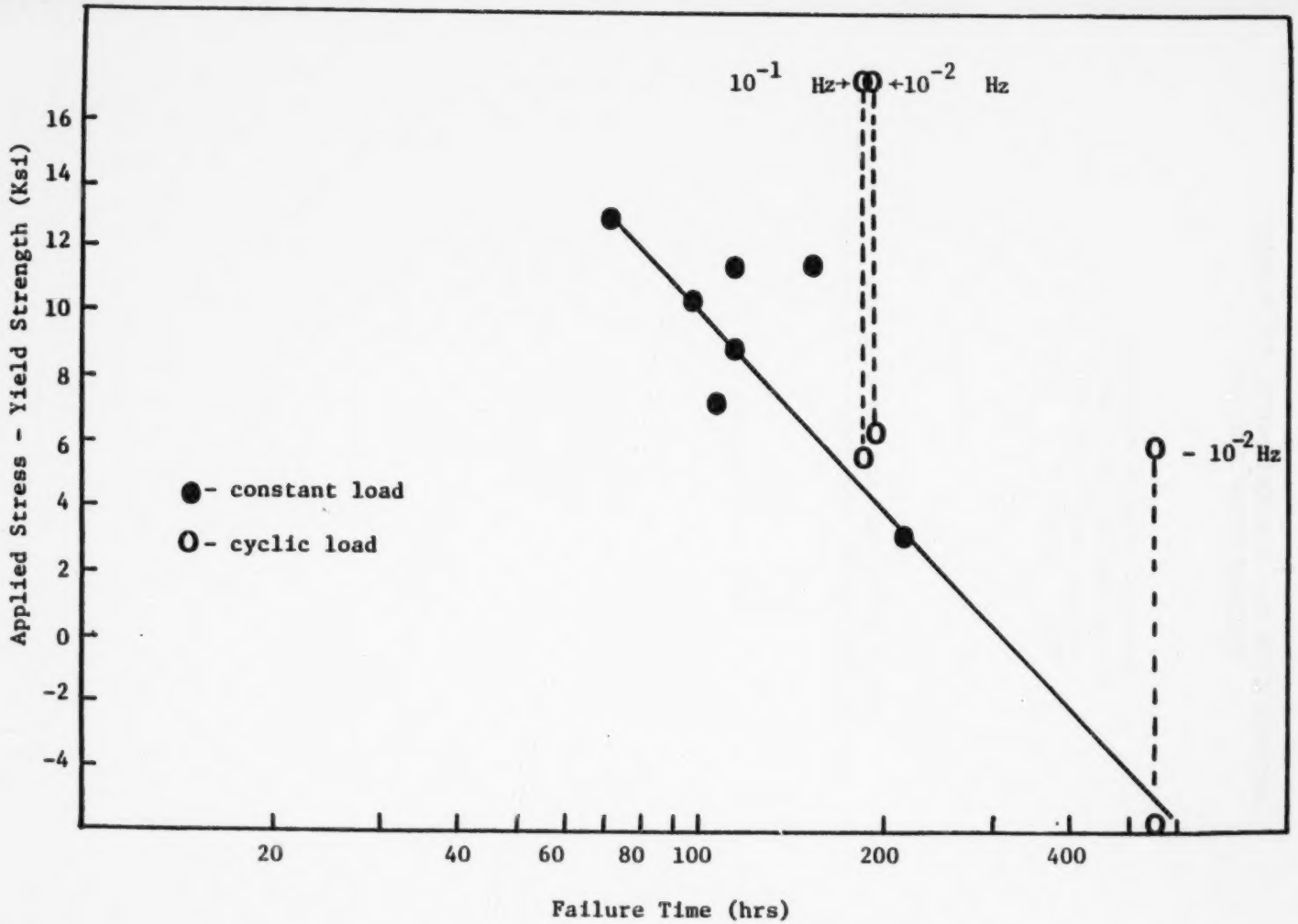


Crack Velocity of .01% Carbon Material Corrected for Initiation Time


SCC INITIATION
CERT .01% C
Extrapolation
290°C = 12.5%
% Strain = k e^(-Q/RT)
k = 7.7 x 10^-5
Q = 6,760 cal/mole
% Strain
20
10
9
8
7
6
5
4
3
2
1
0.5
365°C
345°C
325°C
290°C
1.55
1.65
1.75
1000/°K



Effect of Cold Reduction on Crack Propagation Rates



Comparison of Constant Load and Cyclic Load Tests in 365°C D.I. H_2O

PROGRAM FOR ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS

Materials Science Division
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439

PRESENTATION FOR THE EIGHTH WATER
REACTOR SAFETY RESEARCH INFORMATION MEETING

October 31, 1980

PROGRAM FOR ENVIRONMENTALLY ASSISTED CRACKING IN LIGHT WATER REACTORS

Argonne National Laboratory

Stress-corrosion cracking (SCC) is a generic problem in light water reactors (LWRs). In Boiling Water Reactors (BWRs), the problem has escalated in recent years from cracks in small-diameter (e.g., 4-in.) piping to cracks in the Emergency Core Cooling System and the largest (e.g., 24-in.) primary-system recirculation lines. The cracks, normally located in sensitized stainless steel, are difficult to detect until substantial leakage occurs. A major utility/industry effort is under way to address these problems. To date, most progress has been made in providing additional margin against SCC in new plants through alternate materials and fabrication procedures. For existing plants, the most promising remedy for the prevention of SCC appears to be through lowering the oxygen content of the coolant water, although only preliminary assessments of this approach have been made.

In Pressurized Water Reactors (PWRs), SCC is usually found outside the primary system, since the oxygen level in the primary coolant is kept low through additions of hydrazine and a hydrogen overpressure. It has occurred recently in PWR secondary systems, including the decay-heat-removal system, the spent-fuel-pool piping system, and the steam generator feedwater piping. PWR steam generator tubing has also experienced SCC, and most recently, SCC has occurred in PWR turbine discs.

Detection of SCC through normal ultrasonic in-service inspections is very difficult. As a result, many through-wall cracks have occurred, and increased reliance must be placed on leak-detection systems. Current efforts toward improving in-service inspection are focused on the use of pattern recognition and signal-processing techniques to reduce the dependence on the judgment and experience of the operator. Acoustic emission techniques for leak detection are promising, but require additional research and development.

Although the current understanding of the mechanisms of SCC in LWRs is incomplete, it is generally recognized that three primary factors are required: (1) a susceptible material, (2) a tensile stress, and (3) a corrosive environment. Elimination of all or any one of these conditions can produce immunity to SCC.

Austenitic stainless steels can become susceptible to intergranular stress corrosion cracking (IGSCC) as a result of chromium depletion near the grain boundaries by chromium carbide precipitation in grain boundaries of the heat-affected zones next to welds or heat-treated components. Nucleation of the carbides, even without significant growth at the high welding or heat-treating temperatures, may lead to susceptibility with time at reactor operating temperatures. This latter phenomenon is referred to as low-temperature

sensitization (LTS). Quantitative measures of sensitization are being developed, as well as improved welding techniques to reduce the degree of sensitization. Alternate materials have been proposed that attempt to avoid sensitization through lower carbon content or through alloy additions that promote the formation of carbides not containing chromium. The problem of LTS, however, has not been fully explored for either austenitic stainless steels or alternate materials.

Residual stresses resulting from weld shrinkage together with operating stresses appear to provide the driving force for IGSCC, and fabrication techniques that induce compressive stresses on the inner surface of the weldment seem to increase the resistance to IGSCC. In larger diameter piping there is limited evidence of reduced tensile surface residual stresses and large compressive mid-wall stresses. This may account for the much lower frequency of cracking observed in the larger-diameter lines and for the geometry of the cracks that have occurred.

Several fabrication techniques and alternate materials have been developed which can be used for new construction or repairs which may greatly increase the margin against IGSCC. An extensive testing program is being carried out by industry to qualify these techniques and materials. Full-scale piping tests are being used to develop factors of lifetime improvement for the remedies in comparison with Type 304 stainless steel. The quantitative meaning of these factors is open to question, but the qualitative results are encouraging.

For existing BWRs, the addition of hydrogen to the primary feedwater, along with deaeration during operating transients, is currently under study as a means of preventing SCC in the primary system. The question of potential side effects, however, must be resolved. Other remedies that can be applied to existing plants are based on changing the state of residual stress on the inner surface of a weldment from tensile to compressive. The effectiveness and feasibility of these techniques have not yet been fully established, although extensive research programs to assess them are being carried out in the US and Japan.

Since it is now clear that IGSCC can occur anywhere in the recirculation system of BWRs, including main recirculation lines, it is important to be able to evaluate the margin against failure remaining in lines with shallow cracks. Experimental data on crack propagation rates and residual stress distributions near weldments are being obtained. Analytical models to predict remaining lifetimes and crack shapes are being developed. However, these models are based on linear elastic fracture mechanics (LEFM), and the applicability of LEFM to the analysis of SCC propagation requires further study.

Despite the large investment by industry in various aspects of the SCC problem, only very limited efforts have been devoted to the development of design approaches through a fundamental understanding of the problem, although some recent work in this regard is promising.

Based on the foregoing considerations and an analysis of current efforts by industry, the additional research and development needed by the NRC to develop an independent capability to evaluate current SCC problems in LWRs and their potential solutions is assessed. Six general areas of research are considered: (1) Leak Detection and Nondestructive Evaluation, (2) Analysis of Sensitization, (3) Analysis of Crack Growth, (4) Nonenvironmental Corrective Actions, (5) Environmental Corrective Actions, and (6) Basic Understanding of SCC in LWRs. Assuming the problems do not escalate substantially in the meantime, it is estimated that a five-year period will be required to complete the effort, lagging the much larger efforts of the utility industry by about one year.

ENVIRONMENTALLY ASSISTED

CRACKING IN LWR SYSTEMS

ARGONNE NATIONAL LABORATORY

OBJECTIVE: TO DEVELOP AN INDEPENDENT CAPABILITY FOR THE PREDICTION, DETECTION, AND CONTROL OF SCC IN LWR SYSTEMS, AND FOR THE EVALUATION OF THE TECHNICAL MERITS OF PROPOSED SOLUTIONS.

Fig. 1

I. BACKGROUND

A. BWRs

B. PWRs

II. TECHNICAL ISSUES

III. PROPOSED PROGRAM FOR NRC

Fig. 2

BWRs

1965 • IGSCC IN FURNACE SENSITIZED COMPONENTS.

1974 • IGSCC IN PIPE WELDMENTS (≤ 10-IN. LINES).

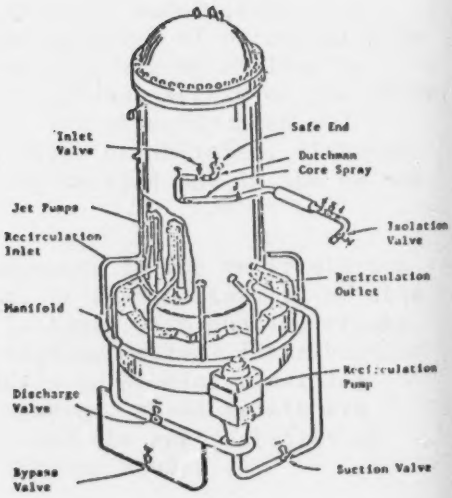
1976 • CORROSION FATIGUE OF FEEDWATER NOZZLES.

1978 • IGSCC IN A 24-IN. PRIMARY RECIRCULATION LINE.

• ALL EIGHT 10-IN. RECIRCULATION-INLET-NOZZLE SAFE ENDS ON DUANE ARNOLD SEVERELY CRACKED.

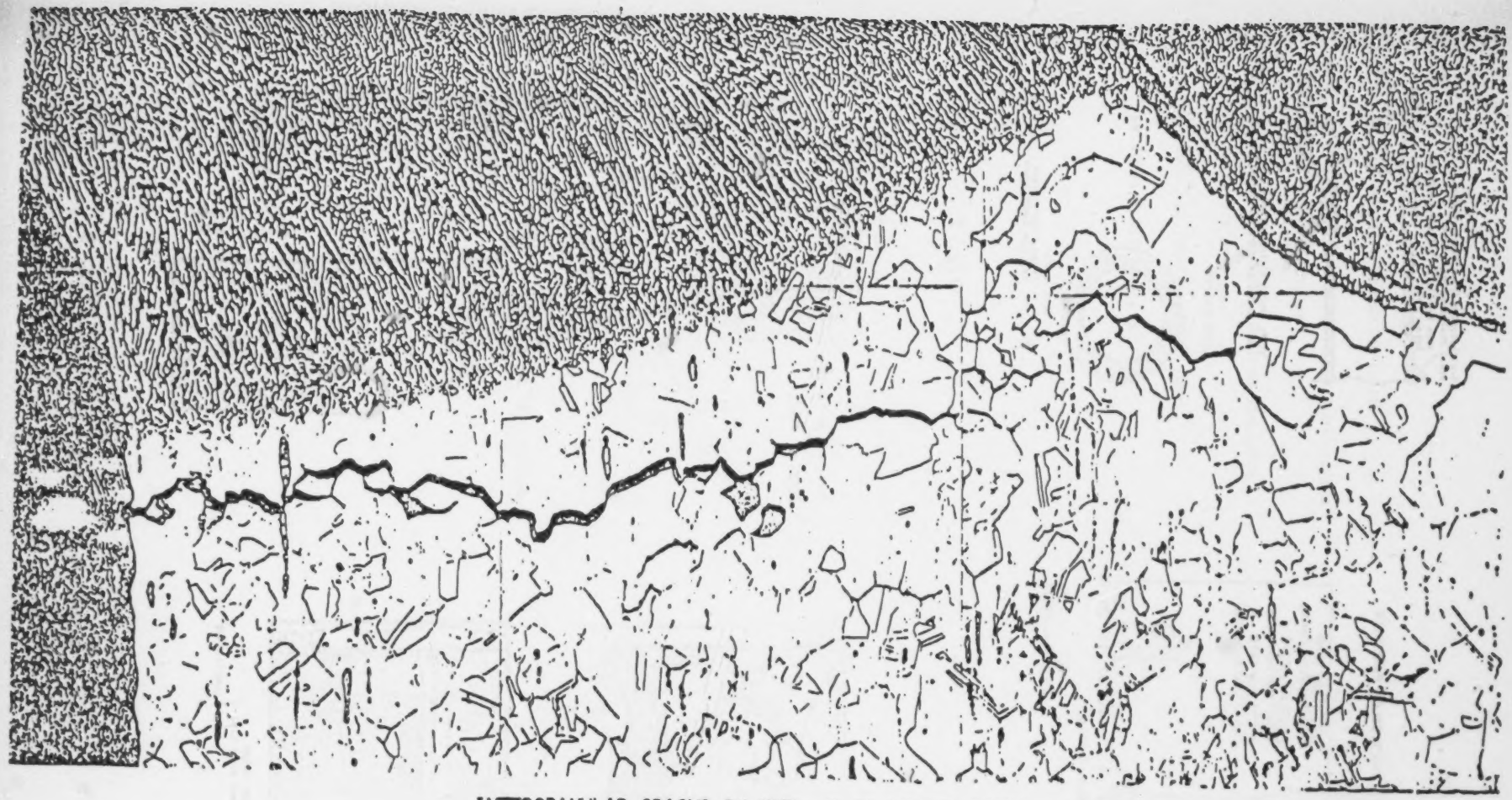
1980 • IGSCC PROPAGATION THROUGH A WELD ON A QUAD CITIES 10-IN. CORE SPRAY LINE.

Fig. 3



BWR PRIMARY SYSTEM PIPING

Fig. 4



INTERGRANULAR CRACKS IN HAZ D-II LOOP A 10K14 MAG 50

(ANL - 1974)

Fig. 5

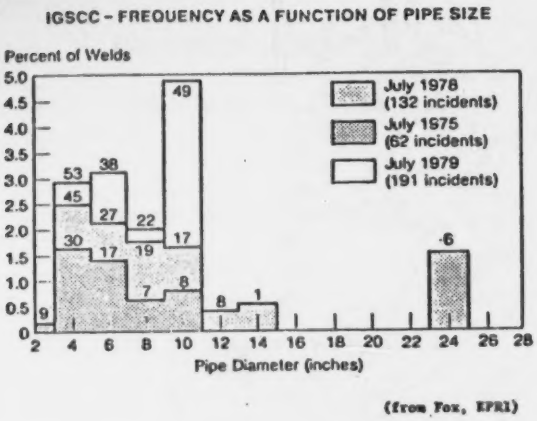


Fig. 6

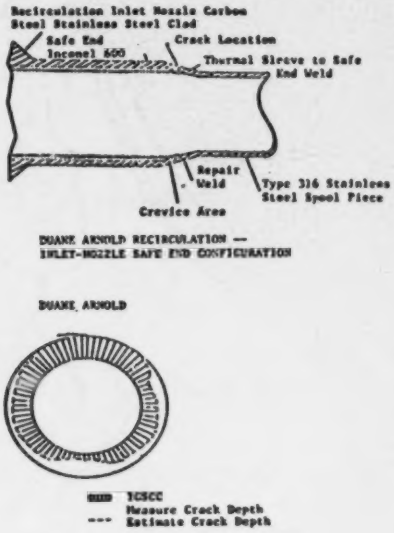


Fig. 7

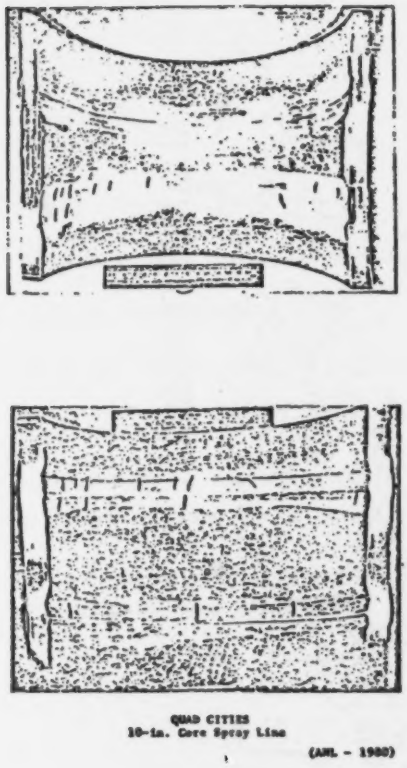


Fig. 8

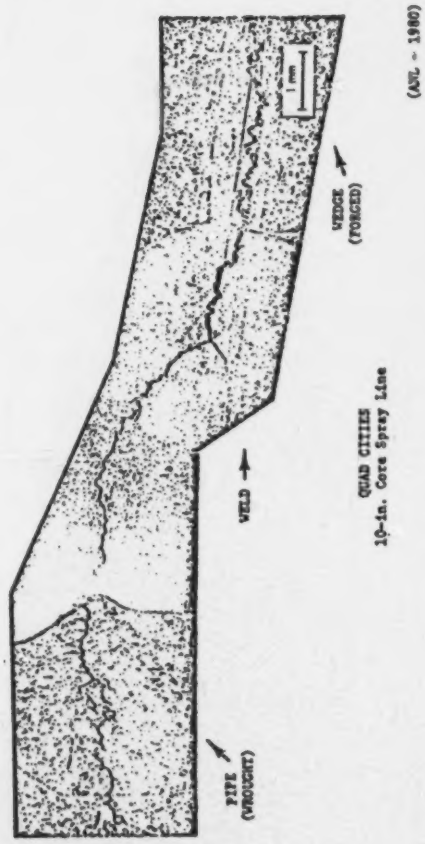


Fig. 9

PWRs

PRE-1974 ● SCC OF STEAM GENERATOR TUBES.

1975 ● DENTING AT STEAM GENERATOR TUBE/SUPPORT PLATE INTERSECTIONS.

1979 ● CORROSION FATIGUE IN STEAM GENERATOR FEEDWATER NOZZLES.

● IGSCC IN BORATED WATER LINES.

● TURBINE DISC CRACKING.

Fig. 10

TUBE WALL DENTING

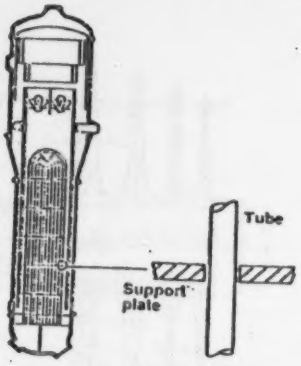


Fig. 11

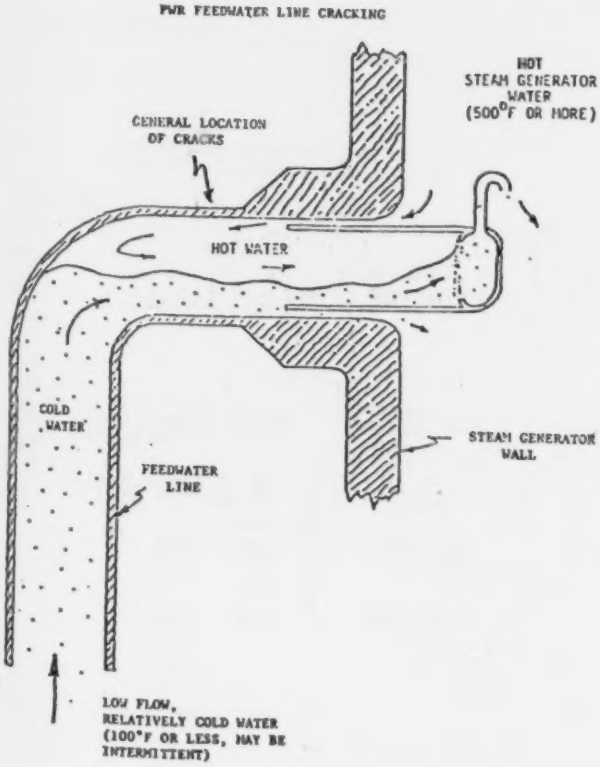
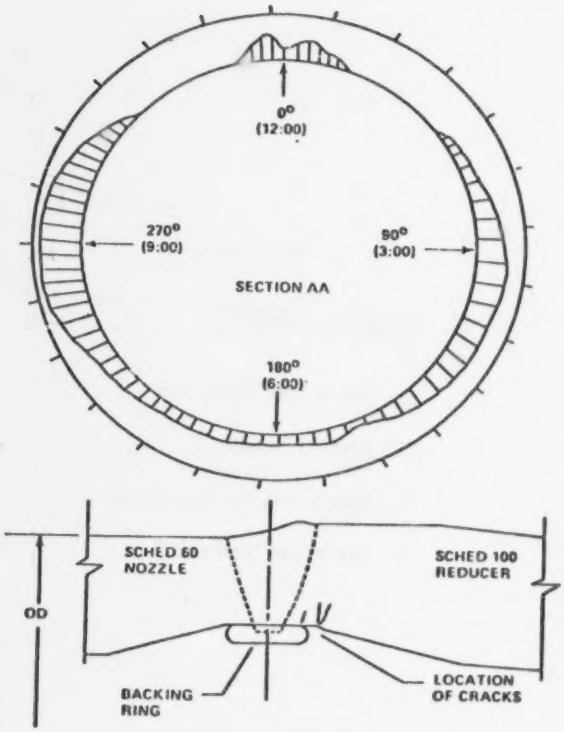
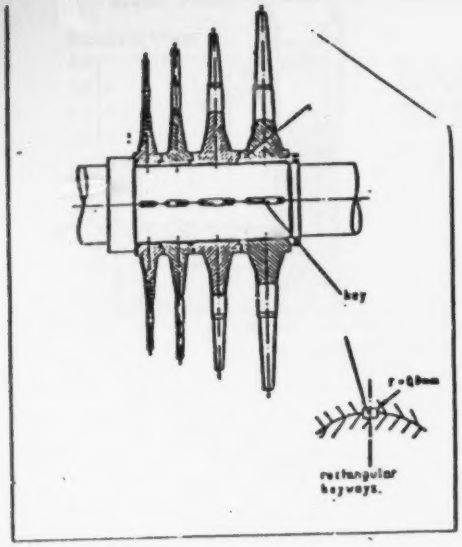


Fig. 12



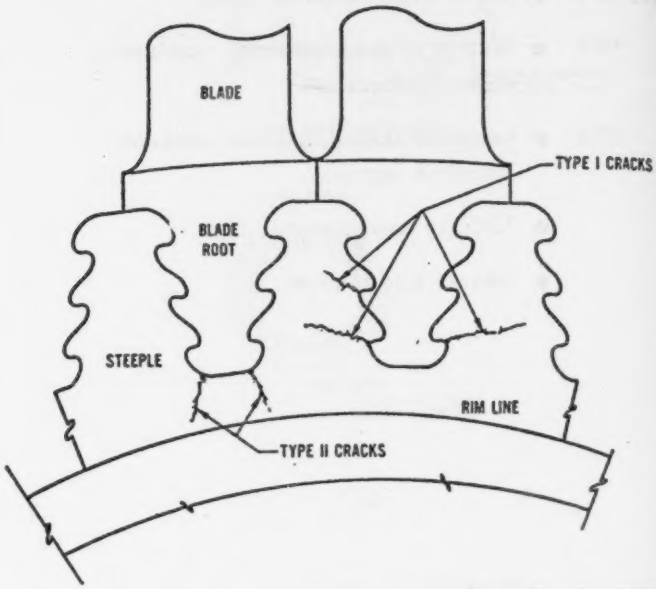
H. B. ROBINSON (1979)

Fig. 13



LP shaft with shrink-fit discs

Fig. 14



(Burghard, SWRI)

Fig. 15

II. Technical Issues

1. NDE & Leak Detection
2. Sensitization
3. Margin for Cracked Pipes
4. Basis for "Remedies"

Fig. 16

DETECTION OF BWR PIPE CRACKS

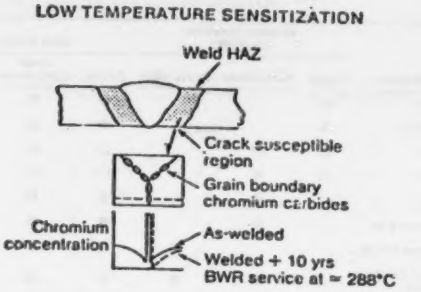
	DETECTED BY:	
	UT	LEAK
PRE-1975	55%	45%
POST-1975	81%	19%

Fig. 17

Results of Examinations of TMI-1 Piping

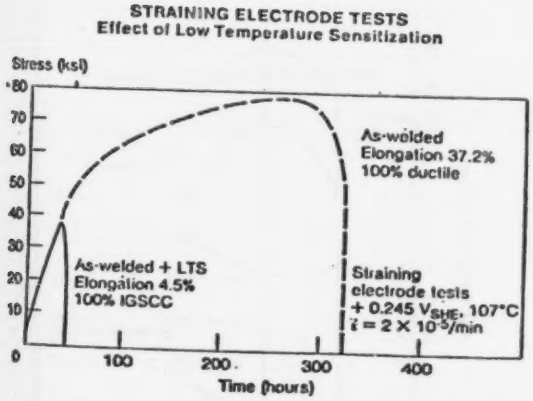
Systems	Total No. of Welds with all UT Inspections Completed	Total No. of UT Indications Using Sensitive Technique	Total No. of UT Indications Evaluated to be Geometric Reflectors	No. of Thru-Wall Leaks
Spent Fuel	521	138	119	5
Decay Heat	277	99	89	1
Bldg. Spray	238	56	48	0
Makeup	693	96	95	0
Core Flood	31	1	0	0

Fig. 18



(Fox, EPRI)

Fig. 19



(Fox, EPRI)

Fig. 20

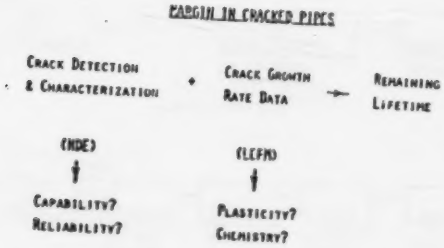


Fig. 21

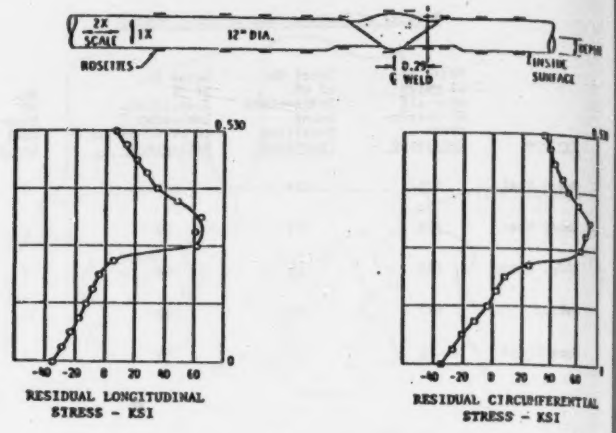
REMEDIES FOR BWR PIPE CRACKING

Remedy	Essential Conditions for IGSCC: Stress	Sensitization	Environment	BWR Status: Existing	Under Construction	Future
IHSI	●			●	●	○
LPHSW	●			●	●	○
HSW	●	○		○	●	○
SHT	○	●		○	●	○
CRC		●		●	●	○
New material		●		●	●	●
Oxygen control			●	●	●	○
Water chemistry control			●	●	●	○

● Primary condition or BWR status influenced
○ Secondary condition or BWR status influenced

(Fox, EPRI)

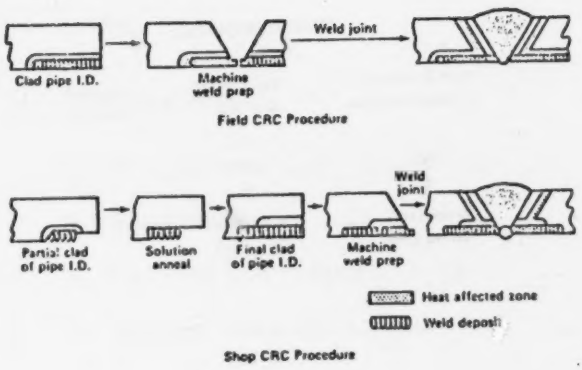
Fig. 22



Last Pass Heat Sink Welding

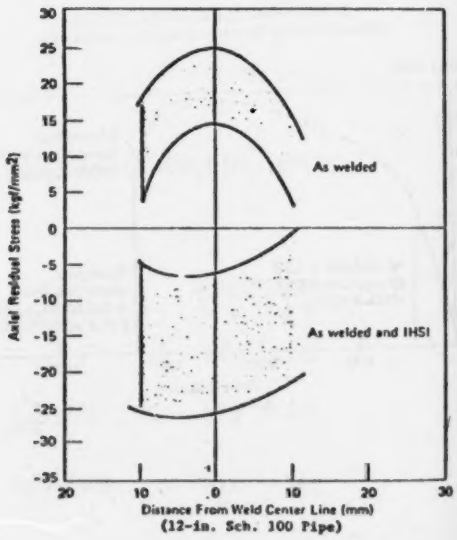
(Fox, EPRI)

Fig. 23



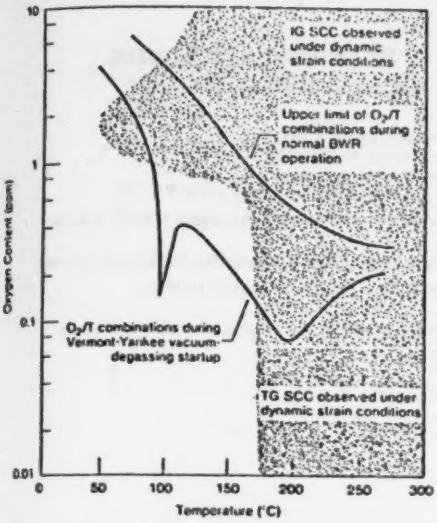
(Fox, EPRI)

Fig. 24



(Fox, EPRI)

Fig. 25



CERT TEST RESULTS

(Ford, G.E.)

Fig. 26

Pipe Test Results for Alternate Alloys

Alloy	Pipes Tested	Pipes Failed by IGSCC	Factor of IGSCC Improvement
Type-304	19	18	1
Type-316	3	3	~1
Type-316 Nuclear Grade	11	0	≥51
Type-304 Nuclear Grade	8	0	≥48
Type-347	9	0	≥42
Type-CF3	8	0	≥51
SA106 Grade B, SA333 Grade 6	6	(Non-IGSCC Failures)	~8-20

Fig. 27

Factors of Improvement in Pipe Tests

Heat Number	Remedy	Factor of Improvement
M7616	SHT	≥89.5
M0063	SHT	≥ 8.7
434970	SHT	≥ 3.0
M7616	CRC Field	6.5
M0063	CRC Field	≥ 7.5
434970	CRC Field	1.9
M7616	CRC Shop	≥88.1
M0063	CRC Shop	≥ 7.7
434970	CRC Shop	≥ 1.9
M7616	HSW 136% of σ_y	25.2
M7616	HSW 136% of σ_y	15.1
M7616	HSW 110% of σ_y	≥ 7.7

Fig. 28

PIPE TEST ISSUES

1. CONSERVATIVE?
   - LOAD HISTORY
   - SENSITIZATION
   - POST-WELD GRINDING ON REFERENCE
2. SHAKEDOWN OF STRESS, "REMEDIES"
3. CONSTANT WATER CHEMISTRY

Fig. 29

ENVIRONMENT ISSUES

BWRs

- Oxygen Level
  - Deaeration
  - Hydrogen Additions
- Minor Impurities?
- Need for In-reactor Tests?

PWRs

- Effect of Thiosulfate
- Effect of Boric Acid
- Turbine Chemistry

Fig. 30

ENVIRONMENTALLY ASSISTED CRACKING IN LWR SYSTEMS

I) Leak Detection & NDE.

II) Analysis of Sensitization.

III) Validation of Fracture Mechanics for SCC Growth.

IV) Evaluation of Nonenvironmental Corrective Actions for SCC.

V) Evaluation of Environmental Corrective Actions for SCC.

VI) Mechanistic Studies.

Fig. 31

I) Leak Detection & NDE

A) Leak Detection

1. Evaluate commercial leak detection devices.
2. Assess capability of acoustic emission surveillance systems for leak detection.
   - a) Effects of Crack Geometry on leak rates and acoustic signals.
   - b) Effect of background noise on detectability.

B) NDE

1. Collect/prepare field calibration standards.
2. Evaluate new UT techniques for IGSCC.
3. Improve probe design and signal analysis to distinguish between SCC and geometrical reflectors.
4. Develop techniques for through-weld inspections.

Fig. 32

II) Analysis of Sensitization

A) Determine kinetics of LTS in Type 304 SS and alternate materials.

1. Effect of temperature.
2. Effect of strain.
3. Effect of alloy chemistry.

B) Effect of "remedies" on LTS.

C) Use of EPR to qualify materials.

Fig. 33

III) Validation of Fracture Mechanics for SCC Growth

a) Heat-to-heat variation of crack growth rates.

b) Determine effect of geometry and loading history on rates.

c) Investigate alternative approaches to LEFM for characterizing SCC growth in Type 304 SS.

Fig. 34

IV) Evaluation of Nonenvironmental Corrective Actions for SCC

a) Evaluate quantitative validity of pipe tests.

1. Load history.

2. Water chemistry.

3. Reference basis.

b) Investigate stability of stress remedies under operating loads.

c) Measure residual stresses in different weld geometries.

Fig. 35

V) Evaluation of Environmental Corrective Actions for SCC

a) Evaluate effects of deaeration and H_2 additions to BWR coolant.

b) Determine role of chloride, thiosulfate, and boric acid on SCC in PWRs.

c) Evaluate need for in-plant testing.

d) Establish meaningful limits on operating chemistry for LWRs.

Fig. 36

VI) Mechanistic Studies

Develop a mechanistic basis for design against SCC.

a) Model development for static-load SCC.

b) Extend model to corrosion-fatigue.

c) Use model to validate and extrapolate the experimental program and to devise design rules.

Fig. 37

IMPROVED EDDY-CURRENT INSPECTION OF STEAM GENERATOR TUBES*

C. V. Dodd

Metals and Ceramics Division
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830

ABSTRACT

We are developing eddy-current inspection techniques to measure wall thickness, defect size, and denting in steam generators. Of particular concern has been the intergranular attack in and just above the crevice region in the tubesheet. We have constructed a microcomputer-controlled eddy-current testing system for examining steam generator tubes from a remote inspection station. The eddy-current instrument measures the magnitude and phase of the coil output voltage at three preselected frequencies. The coil design and operating frequencies are chosen by preliminary computer calculations, as are the mathematical formulae used to calculate the tube properties from the eddy-current readings.

The inspection problem is complicated by the fact that the steam generator tubes are held in place by ferromagnetic tubesheets and support plates, which strongly affect the eddy-current signals. In order to distinguish the relatively small defect signals from the large extraneous signals caused by other sources, it is necessary to use a rather sophisticated calculation scheme, which consists of

*Research sponsored by the Office of Nuclear Regulatory Research, Nuclear Regulatory Commission, under Interagency Agreement DOE 40-551-75 with the Department of Energy under contract W-7405-eng-26 with the Union Carbide Corporation.

By acceptance of this article, the publisher or recipient acknowledges the U.S. Government's right to retain a nonexclusive, royalty-free license in and to any copyright covering the article

expressing the desired properties, such as defect size, as polynomial expansions in terms of the eddy-current readings. The polynomial coefficients are chosen by a linear least-squares fitting program on a minicomputer and then stored in the memory of the on-board microcomputer in the eddy-current instrument.

The microcomputer is also programmed to drive the probe pusher-puller, record the eddy-current readings, calculate the properties of interest, and periodically recalibrate the eddy-current instrument. A digital tape recording is also made of the unprocessed eddy-current readings so that they can be postprocessed in any manner desired.

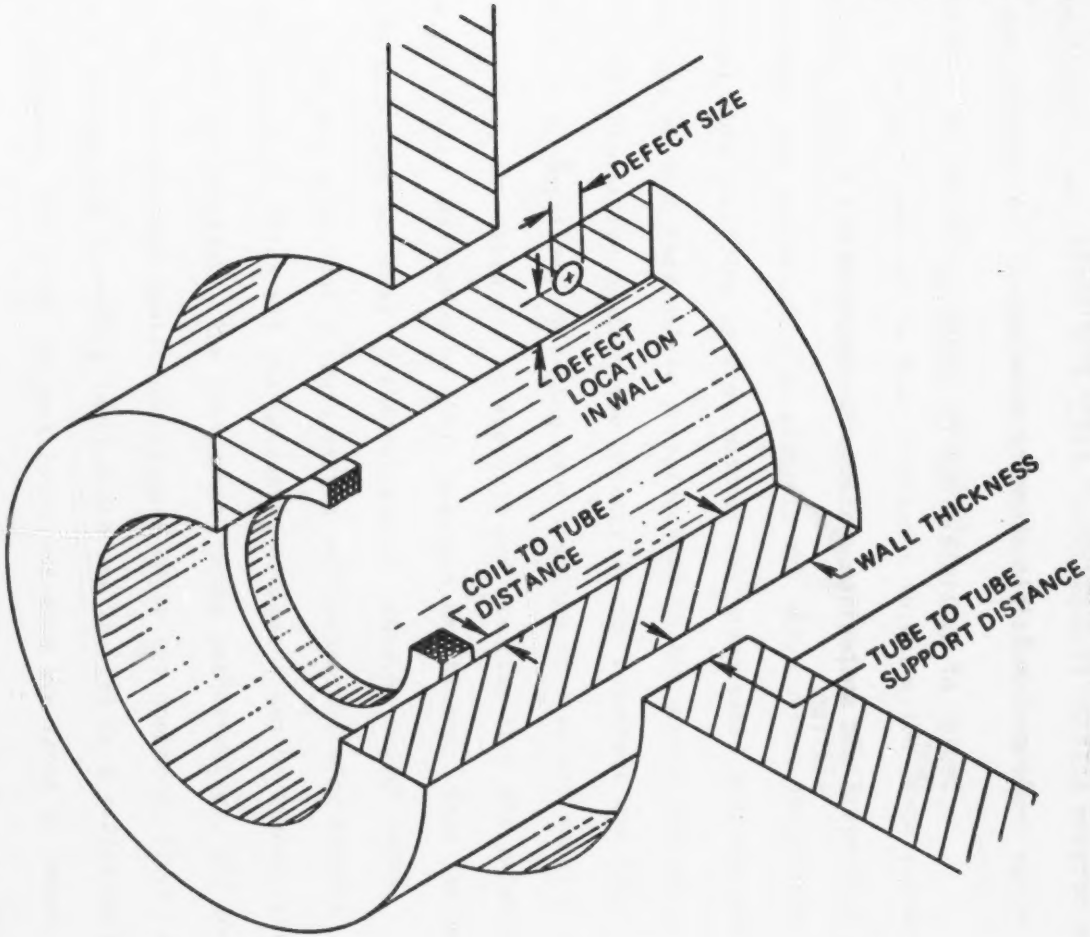
Discrimination against the effects of the tubesheet and tube supports is accomplished by taking many readings of standard samples with and without the tubesheet or supports and ensuring that the polynomial is relatively unaffected by their presence or absence. This can be done with good accuracy if the polynomial contains enough terms.

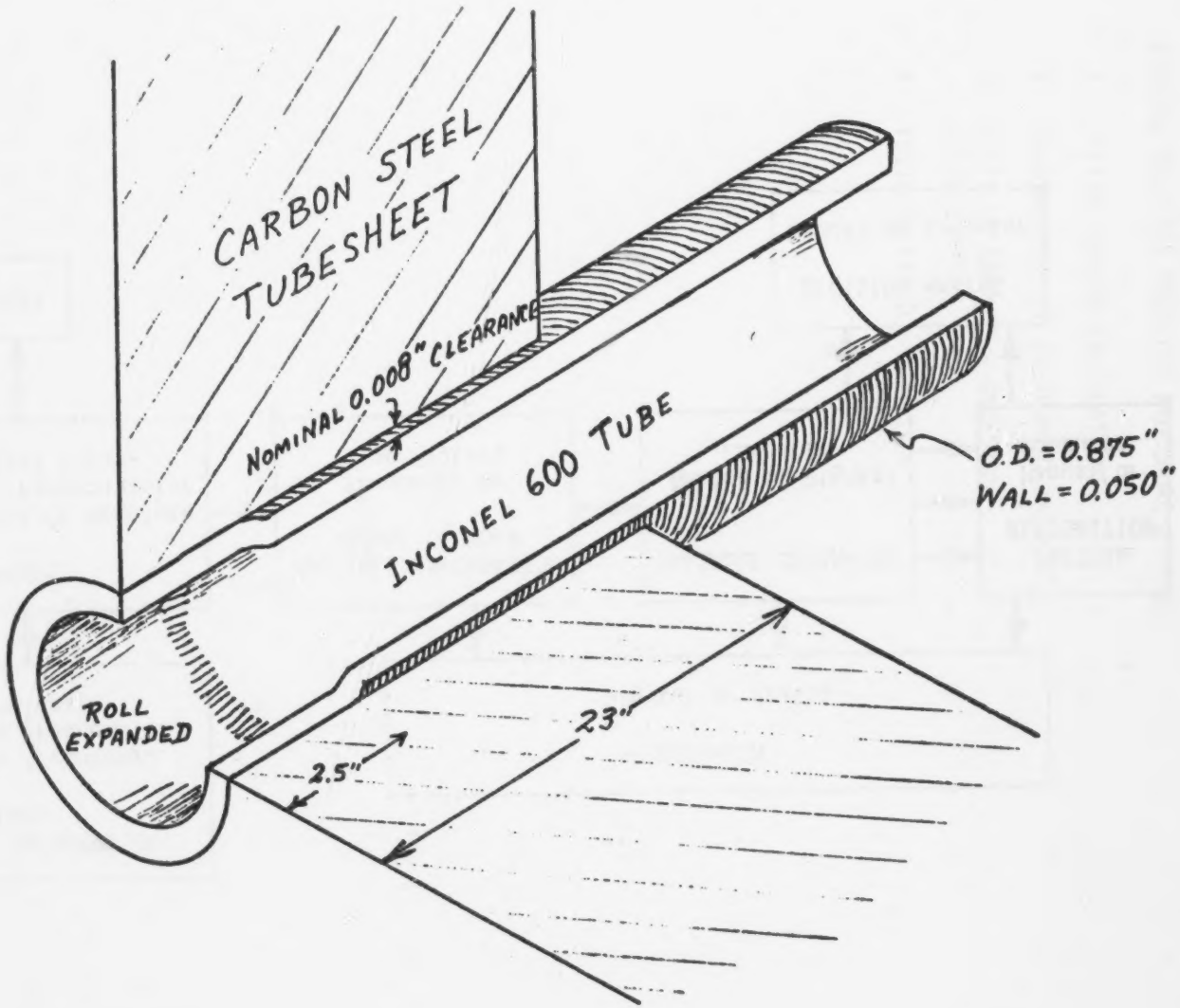
We have outfitted a truck with a self-contained eddy-current laboratory, in which all the necessary equipment can be transported to the reactor location. The instrument, probe pusher-puller, and positioner are operated inside containment while the operator, truck, and data storage and recording instruments are located in a radiation-free environment. The probe is positioned and data are taken by remote control. The pertinent tube properties are displayed instantaneously on a CRT terminal and a strip-chart recorder while the raw data are digitally recorded on tape for later use.

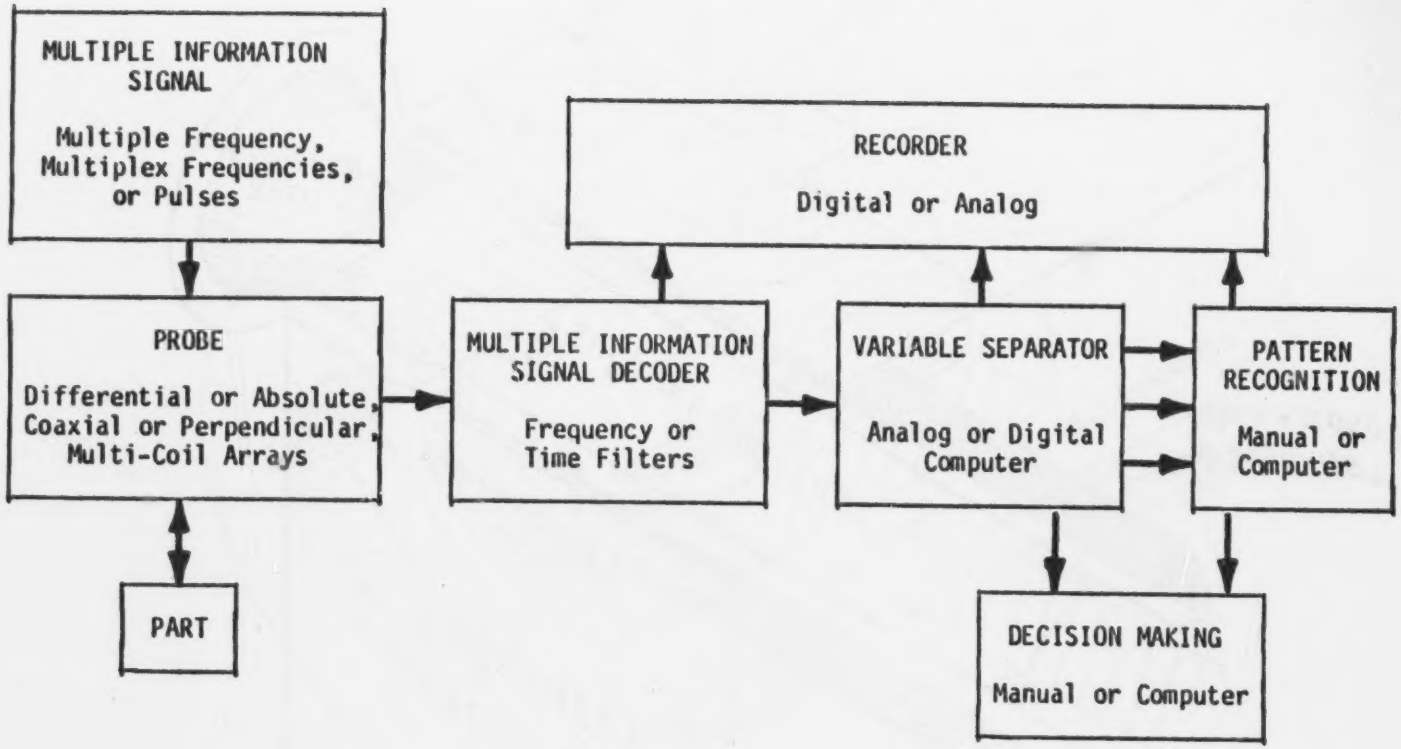
IMPROVED EDDY-CURRENT INSPECTION OF STEAM GENERATOR TUBES

- Inspection problems in steam generators
- General multiple-property solution
- Design of multiple-property system at ORNL
- Field inspection system with remote data station

ORNL-DWG 79-8031




CARBON STEEL
TUBESHEET
NOMINAL 0.008" CLEARANCE
INCONEL 600 TUBE
O.D. = 0.875"
WALL = 0.050"
ROLL
EXPANDED
2.5"
23"


MULTIPLE INFORMATION SIGNAL
Multiple Frequency, Multiplex Frequencies, or Pulses
PROBE
Differential or Absolute, Coaxial or Perpendicular, Multi-Coil Arrays
PART
MULTIPLE INFORMATION SIGNAL DECODER
Frequency or Time Filters
RECORDER
Digital or Analog
VARIABLE SEPARATOR
Analog or Digital Computer
PATTERN RECOGNITION
Manual or Computer
DECISION MAKING
Manual or Computer

GENERAL MULTIPLE PROPERTY PROCEDURE AT ORNL

1. INITIAL DESIGN CALCULATIONS
2. CONSTRUCTION OF INSTRUMENT AND COIL
3. TESTS ON SAMPLES - REDESIGN CALCULATIONS
4. TESTS IN FIELD

INITIAL DESIGN CALCULATIONS

1. CALCULATE INSTRUMENT READINGS FOR MANY DIFFERENT TEST PROPERTIES. $R = f(P)$.

2. DO A LEAST SQUARES FIT OF VARIOUS NONLINEAR COMBINATIONS OF INSTRUMENT READINGS TO PROPERTIES.

$$P_{11} = C_0 + C_1R_{11} + C_2R_{12} + \ldots + C_NR_{1N} \cdot$$

$$P_{12} = C_0 + C_1R_{21} + C_2R_{22} + \ldots + C_NR_{2N} \cdot$$

$$\vdots$$

$$P_{1M} = C_0 + C_1R_{M1} + C_2R_{M2} + \ldots + C_NR_{MN} \cdot$$

LET $P_{ij} - (C_0 + C_1R_{j1} + \ldots + C_NR_{jN}) = \varepsilon_j$.

DETERMINE $C_0, C_1, C_2, \ldots, C_N$ TO MINIMIZE $\sum_{j=1}^{M} \varepsilon_j^2$.

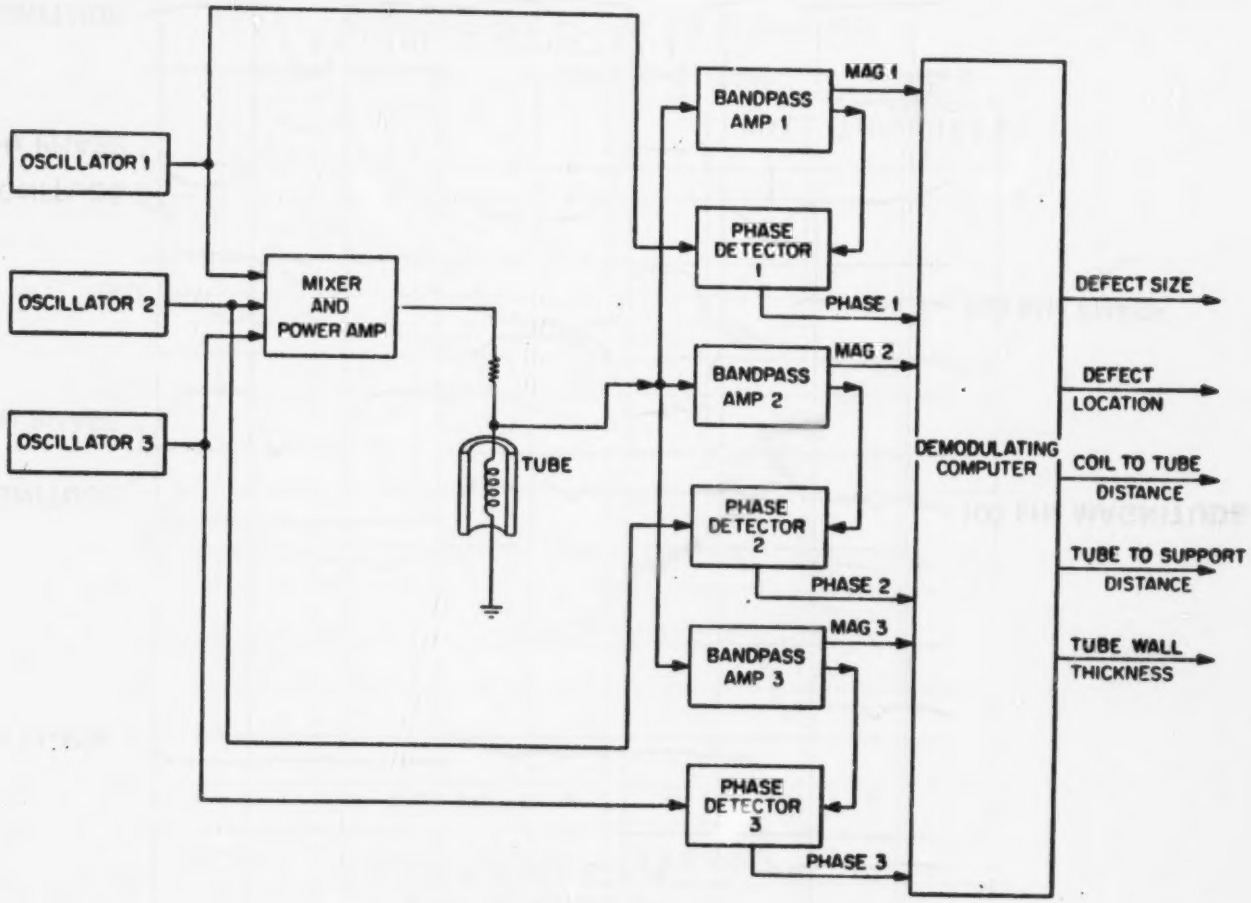
THE R_{jk} CAN BE FUNCTIONS OR POWERS OF THE READINGS:

$$\text{THICK} = C_0 + C_1(\ln M_1) + C_2(\ln M_1)^2 + C_3(Ph_1) + \ldots$$

3. CALCULATE THE ERRORS DUE TO LACK OF FIT AND INSTRUMENT DRIFT.

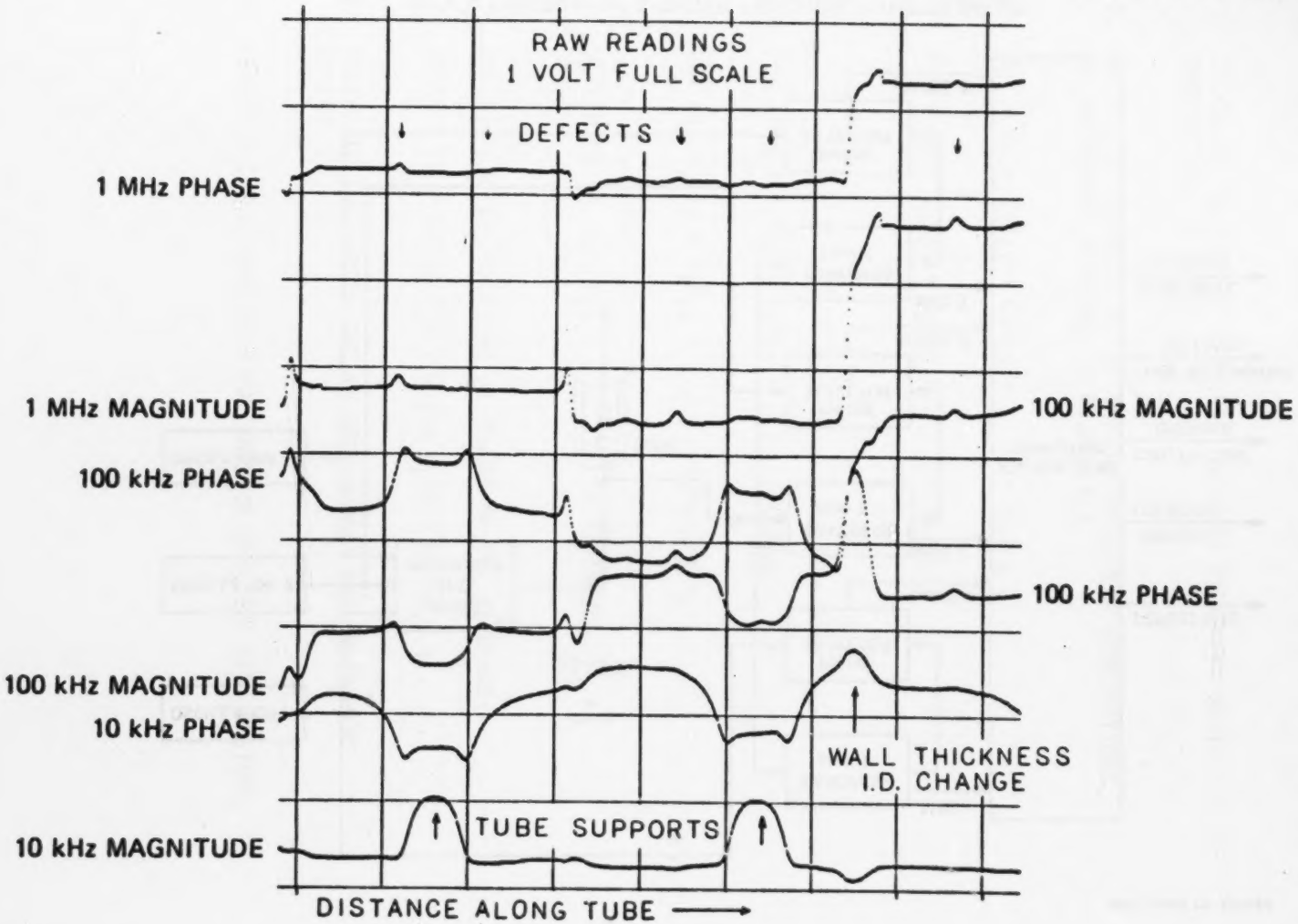
4. REPEAT STEPS 1 THROUGH 3 FOR VARIOUS COIL AND TEST DESIGNS.

ORNL-DWG 78-16168R

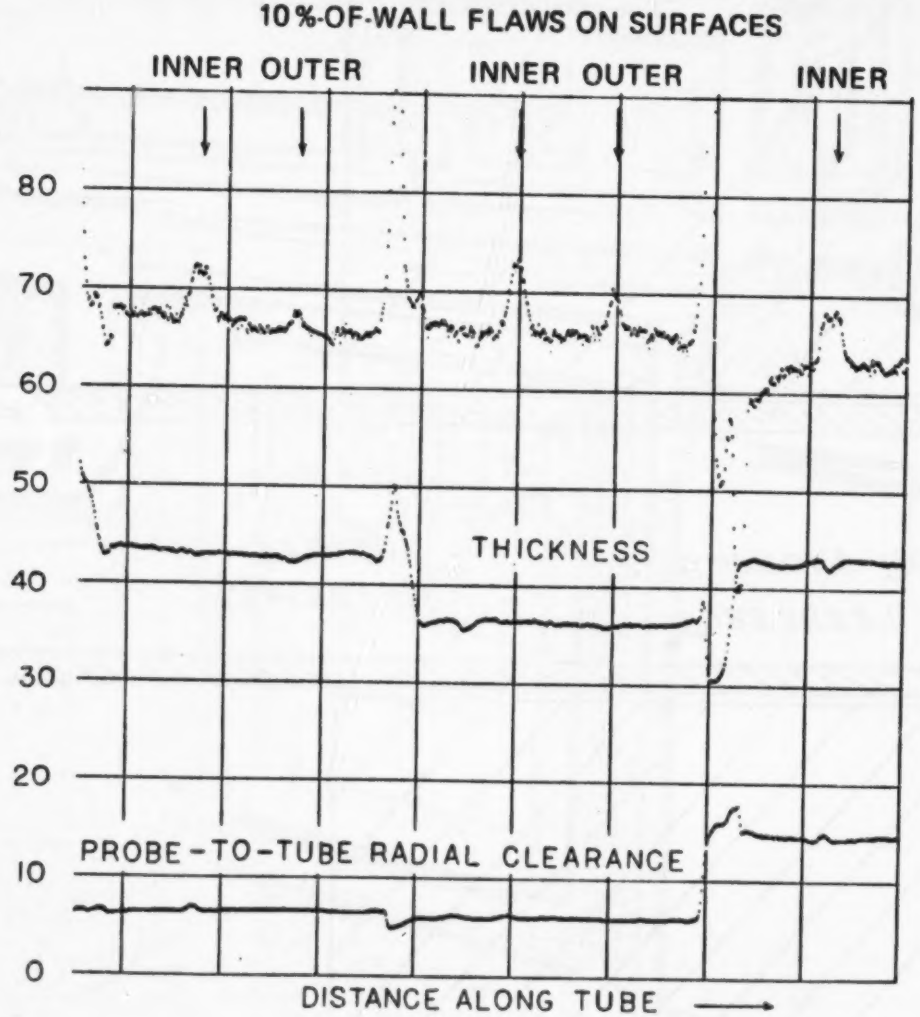


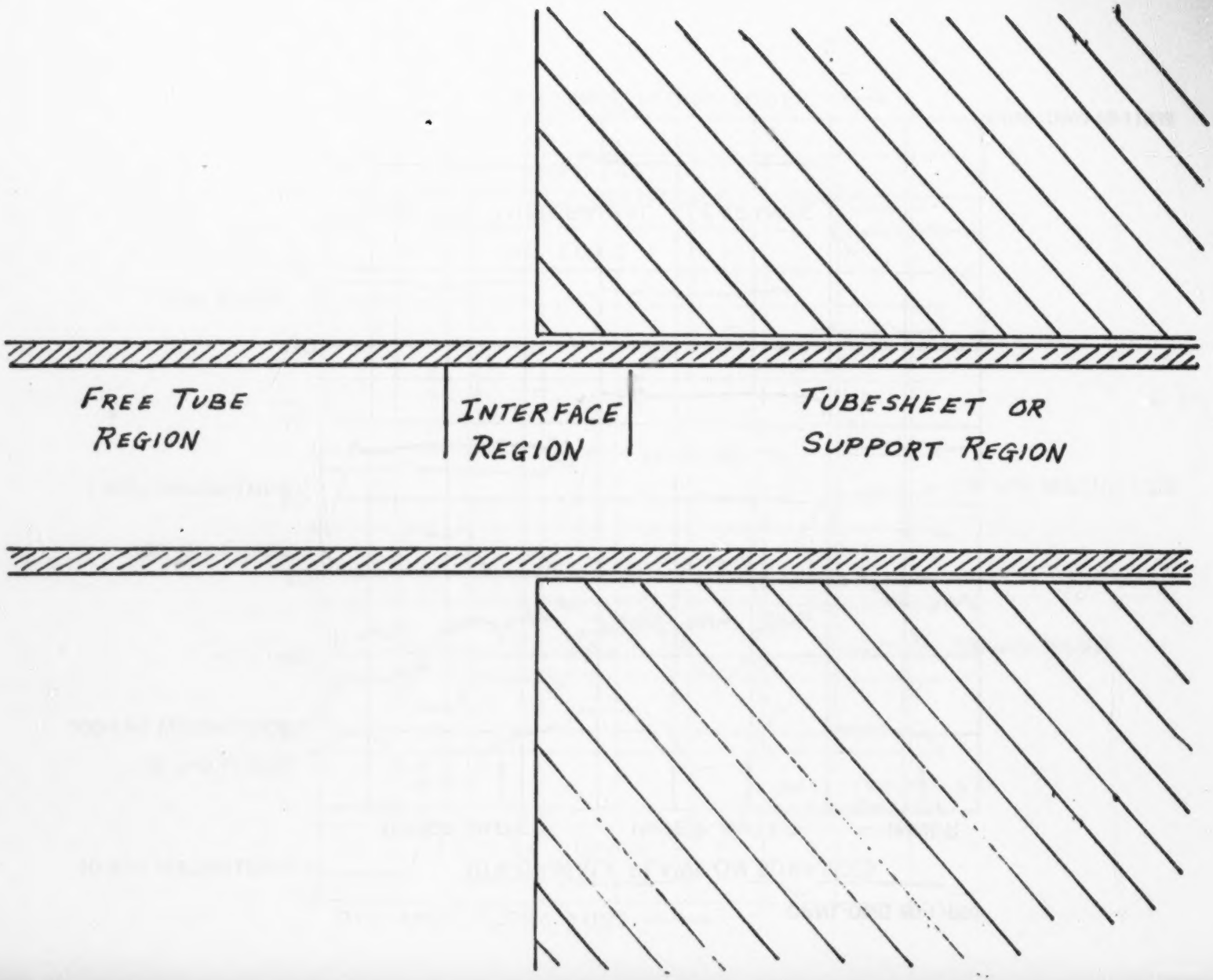
BLOCK DIAGRAM OF A THREE FREQUENCY INSTRUMENT

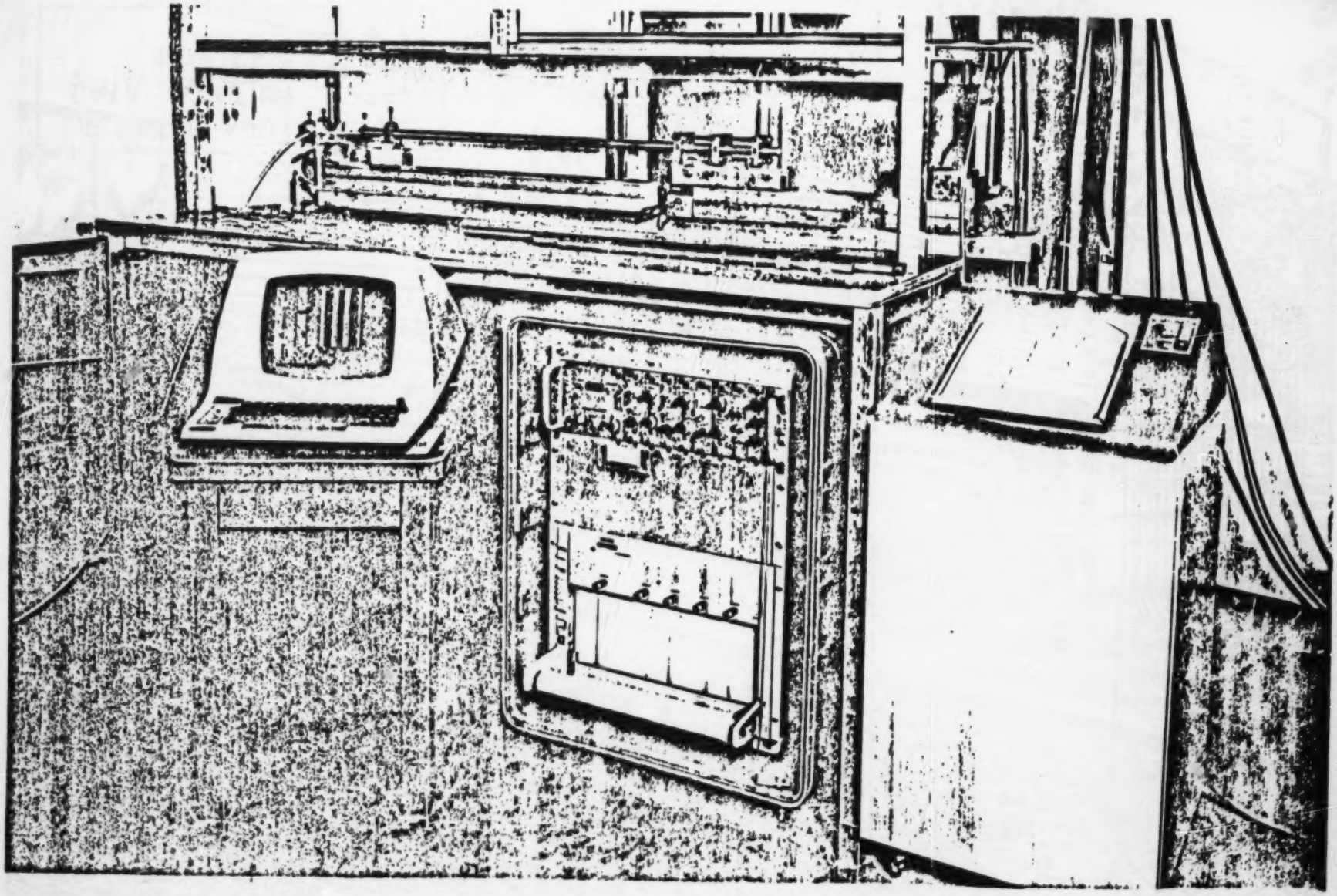
ORNL-DWG 80-11989

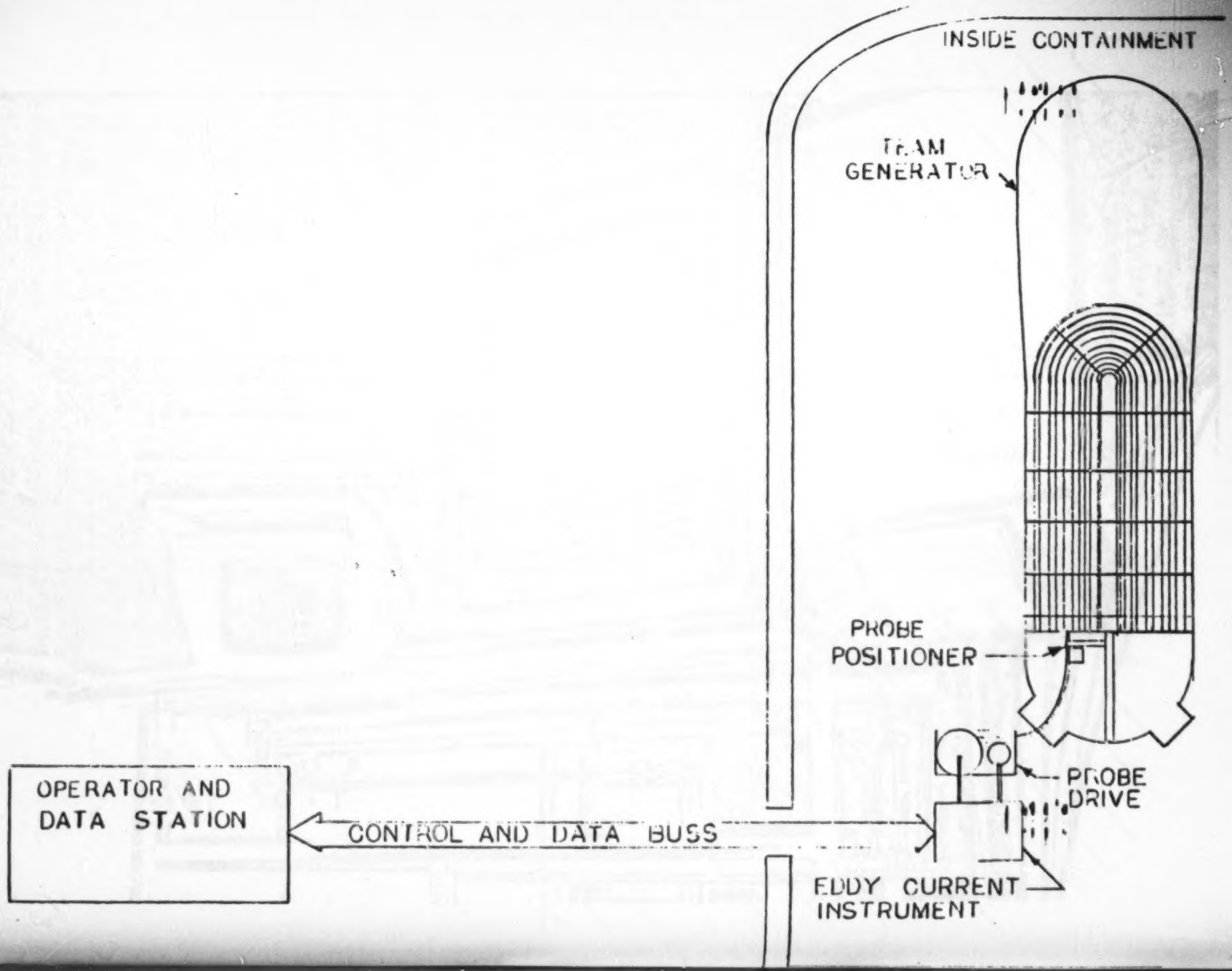


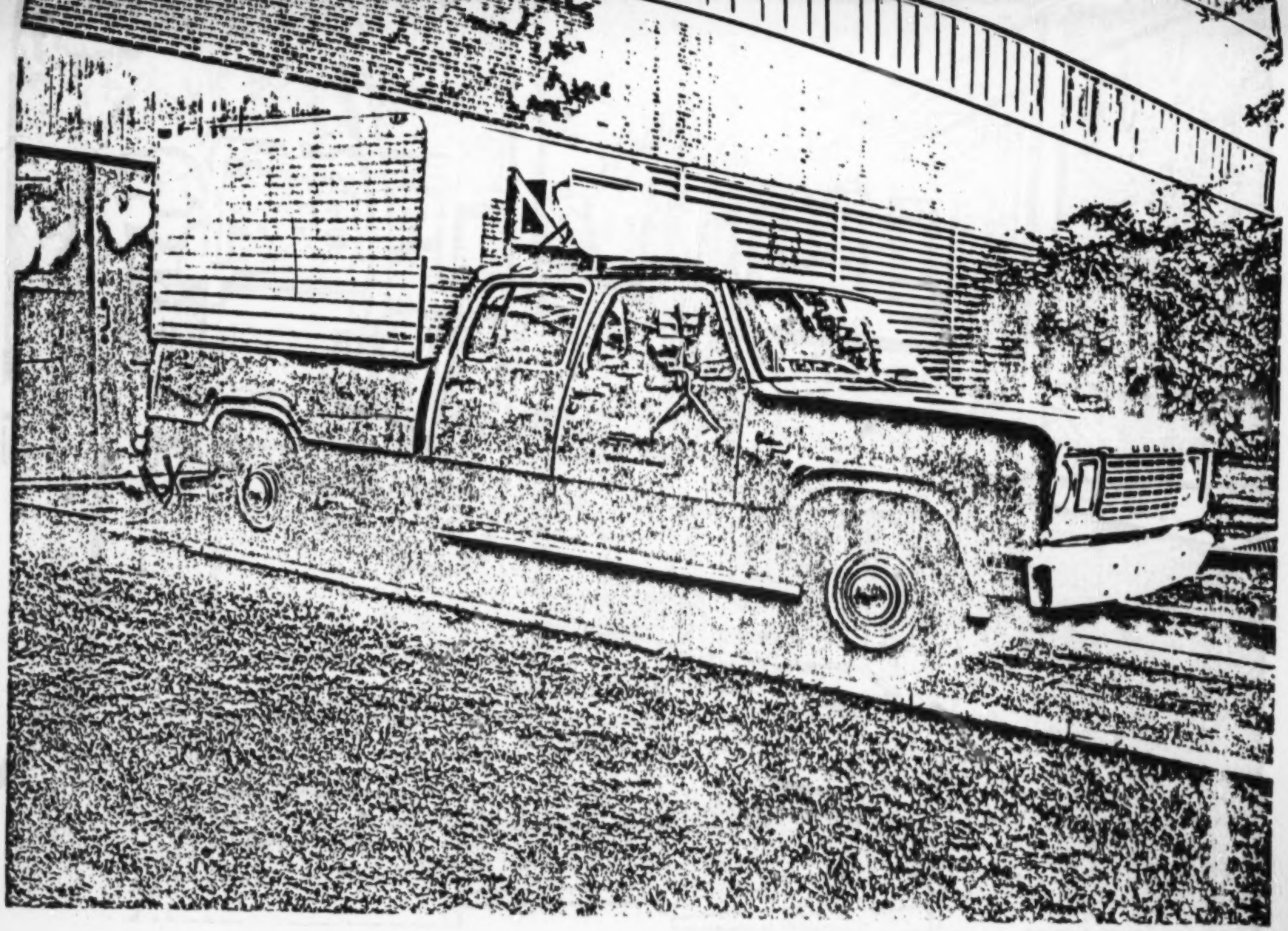
ORNL-DWG 80-11990




FREE TUBE
REGION
INTERFACE
REGION
TUBESHEET OR
SUPPORT REGION




INSIDE CONTAINMENT
STEAM GENERATOR
PROBE POSITIONER
PROBE DRIVE
OPERATOR AND DATA STATION
CONTROL AND DATA BUSS
EDDY CURRENT INSTRUMENT



RELIABILITY OF IN-SERVICE ULTRASONIC FLAW DETECTION FOR PRIMARY PIPING SYSTEMS

F. L. Becker

October 1980

Presented at the Eighth Water Reactor Safety
Research Information Meeting
Gaithersburg, Maryland
October 27-31, 1980

Work Supported by
the U.S. Nuclear Regulatory Commission
under a Related Services Agreement
with the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830

Pacific Northwest Laboratory
Richland, Washington 99352

RELIABILITY OF IN-SERVICE ULTRASONIC FLAW DETECTION FOR PRIMARY PIPING SYSTEMS

F. L. Becker
Pacific Northwest Laboratory (PNL)
Operated by Battelle

INTRODUCTION

This report summarizes the second year's activities on a five-year program which is being conducted by PNL on behalf of the U.S. Nuclear Regulatory Commission. The objectives of this program are shown in Figure 1. The safety and licensing issues addressed by this program are shown in Figure 2. The first year of the program was directed toward identification and quantification of inspection uncertainties which are likely to occur during in-service inspection of primary system piping. These activities were continued during the second year of the program as well as the initiation of a Round Robin inspection program to determine the effectiveness of current inspection practices. The results of the program are described below.

PROGRAM RESULTS

Discussion of the program results is divided into two areas: Inspection Effectiveness and the Round Robin Inspection Program.

INSPECTION EFFECTIVENESS

The effectiveness of an ultrasonic inspection is dependent on many factors. A key parameter is the sensitivity as established by the calibration standard. Other factors include: the flaw characteristics, i.e., tightness, roughness and orientation; instrument and search unit characteristics; as well as variability of the operators and equipment. These other factors and their influences on inspection reliability are discussed in the Phase I Report[1] for this program.

Inspection sensitivity for primary piping system inspections is established by the ASME Boiler and Pressure Vessel Code Section XI, "Rules

for Inservice Inspection of Nuclear Power Plant Components." Applicable revisions include the 1974 Edition through Summer of 75 and the 1977 Edition through Summer of 78. Major changes in the 77 Code include reduction of the inspection volume to the lower one-third of the pipe thickness, introduction of allowable flaw size tables (IWB-3514) and the use of notches as opposed to side drilled holes as calibration reflectors. Reduction of the inspection volume reduces the number of geometric indications (from the weld crown) which must be recorded, while maintaining coverage of the area where in-service flaws will most likely occur. The allowable flaw sizes, listed in IWB-3514, are based on conservative linear elastic fracture mechanics.[2] The notch reflectors specified in the 77 Code are more appropriate than side drilled holes as they more closely approximate in-service defects (surface connected cracks). However, the adequacy of the inspection sensitivity produced by the notch is in question.

In order to evaluate the differences in inspection sensitivity produced by side drilled holes (SDH), 74 Code, and notches, 77 Code, measurements were performed on 34 piping calibration samples. The samples ranged from 4 to 30 inches in diameter and from 0.237 to 2.343 inch nominal thickness. The measurements were performed at half, full and three-halves Vee path, using .25, .5 and 1.0 inch diameter 2.25 MHz shear wave search units at 45° and 60° incidence angles. The differences in inspection sensitivity, for 45° shear waves, are shown in Figure 3. The measured data is the average of approximately 500 measurements along with the $\pm 2\sigma$ error bars ($\sigma \approx 2$ dB). The curve labeled "Theory" corresponds to calculated differences according to Werneyer, et al.[3] The measured and theoretical curves are in substantial agreement. No statistically significant difference in sensitivity was measured for 60° shear wave inspection.

The reduction in sensitivity resulting from calibration according to the 77 Code is significant. A flaw which yields a 100% of DAC reflection according to the 74 Code can be as low as 16 dB below DAC for calibration according to the 77 Code. The calibration notch, described in the 1977 Code, is suitable as a calibration reflector. The inspection sensitivity

produced by the notch varies with search unit selection and is not sufficient to insure detection of even ideal reflectors which are unacceptably large. In addition, the specified reporting level (100% DAC) does not consider the loss in signal amplitude due to flaw characteristics such as tightness, roughness or orientation. Investigations are currently in progress to define a more appropriate calibration reflector as well as the recording and reporting levels. At present the most appropriate calibration reflector appears to be a semicircular notch of sufficiently small size to assure detection of unacceptably large flaws.

ROUND ROBIN INSPECTION PROGRAM

Preparations for a Round Robin inspection program are in progress. The objective of this program is to define the reliability and effectiveness of in-service inspection procedures. Procedures based on minimum Code requirements, as practiced field procedures and recommended improved procedures, will be tested. A second objective is to define the source and magnitude of inspection uncertainties. Towards this end the manual inspection data will be automatically recorded for later analysis. Teams from six industrial ISI organizations will be used to collect the data. It is expected that it will require approximately three weeks for each team to complete the test matrix.

The test matrix is shown in Figure 4. The test matrix consists of: two procedures Code (as applied in the field) and improved; four material/flaw type combinations; two inspection environments (laboratory and difficult); and two access conditions (near and far side). The effectiveness of minimum Code requirements will be measured by evaluating recorded data from the field procedures.

The size of each experiment is dictated by the confidence level which is required. The major experiments are the Code procedures under difficult conditions for near side access and are treated by the full ANOVA experiments of 25 observations. The results of the other experiments are compared to the major experiments and require fewer observations to provide meaningful measures of the differences.

The two types of defects which will be included in the Round Robin are thermal fatigue and intergranular stress corrosion cracks (IGSCC). A schematic diagram of the thermal fatigue process is shown in Figure 5. The thermal fatigue process was chosen as it is adaptable to components of non-uniform geometry, and the cracks are both rough and tight. The depths of the cracks are controllable within satisfactory limits. An example of the type of crack produced is shown in Figure 6.

Intergranular stress corrosion cracks are produced by the graphite wool technique. In this technique the area to be cracked is covered with graphite wool and exposed to 550°F water (8 ppm oxygen) for approximately 600 hours (to produce a 20% through wall crack). The depth and character of the cracks depend on length of exposure, degree of sensitization and residual stress distributions. The cracks produced by this technique are more typical of those which have occured in-service than those produced by other laboratory processes. An example of the type of IGSCC cracks produced is shown in Figure 7.

With the development of these two crack growing techniques, it is now practical to produce cracked samples to be used for the Round Robin and for demonstration of procedure and personnel qualifications.

REFERENCES

1. Becker, F. L., et. al., Investigation of NDE Reliability and Fracture Mechanics, Phase I Report, NUREG/CR-169b, PNL-3469, prepared for the Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington, September 1980
2. Maccary, R. R., Nondestructive Examination Acceptance Standards, Technical Basis and Development of Boiler and Pressure Vessel Code, ASME Section XI, Division 1, NP-1406-SR, Electric Power Research Institute, May 1980
3. Werneyer, R., Walte and Klein, M., Mathematisches Modell zur Fehlerrekonstruktion bei der Ultraschall - Impulsecho und Tandumprufung und Ergebnisse aus Modellversuchen, Institut fur zerstorungsfreie Prufverfahren (IZFP), Saarbruken, West Germany, Report No. 770212-TW, 1977, p. 6

PUBLICATIONS

Investigation of NDE Reliability and Fracture Mechanics, Phase I Report, F. L. Becker, et al., NUREG/CR-169b, PNL-3469, September 1980.

Quarterly Report - July 1-September 30, 1979, F. L. Becker, et al., NUREG/CR-1009, PNL-3040-3, March 1980.

Quarterly Report - October 1-December 31, 1979, F. L. Becker, et. al., NUREG/CR-1349, PNL-3040-4, August 1980.

Quarterly Report - January 1-March 31, 1980, F. L. Becker, et al., NUREG/CR-1454, PNL-3380-1, August 1980.

Quarterly Report - April 1-June 30, 1980, F. L. Becker, et al., NUREG/CR-1762, PNL-3380-2, November 1980 (anticipated).

Quarterly Report - July 1-September 30, 1980, F. L. Becker, et al., NUREG/CR-?, PNL-3380-3, December 1980 (anticiapted).

Evaluation of Flaw Characteristics and Their Influence on Inservice Inspection Reliability, F. L. Becker, Specialist Meeting on Reliability of Ultrasonic Inspection of Austenitic Stainless Steel Components, May 30, 1980, to be published.

NDE/FM

OBJECTIVES:

1. DETERMINE THE RELIABILITY OF ULTRASONIC INSERVICE INSPECTION (ISI) PERFORMED ON COMMERCIAL, LIGHT WATER REACTOR PRIMARY PIPING SYSTEMS

2. USING FRACTURE MECHANICS ANALYSIS, DETERMINE THE IMPACT OF NDE UNRELIABILITY ON SYSTEM SAFETY AND DETERMINE THE LEVEL OF INSPECTION RELIABILITY REQUIRED TO ASSURE A SUITABLY LOW FAILURE PROBABILITY.

3. EVALUATE THE DEGREE OF RELIABILITY IMPROVEMENT WHICH COULD BE ACHIEVED USING IMPROVED AND ADVANCED NDE TECHNIQUES.

4. BASED ON MATERIAL, SERVICE AND NDE UNCERTAINTIES, FORMULATE RECOMMENDED REVISIONS TO ASME SECTION XI AND REGULATORY REQUIREMENTS NEEDED TO ASSURE SUITABLY LOW FAILURE PROBABILITIES.

Figure 1

NDE/FM LICENSING AND SAFETY ISSUES

I EFFECTIVENESS AND RELIABILITY OF INSERVICE INSPECTION

II ADEQUACY OF CURRENT REQUIREMENTS

- INSPECTION TECHNIQUE
- EXTENT AND FREQUENCY OF INSPECTION
- ALLOWABLE FLAW SIZES AND REQUIRED PROBABILITY OF DETECTION

III RECOMMENDATIONS FOR IMPROVED INSPECTION REQUIREMENTS

Figure 2

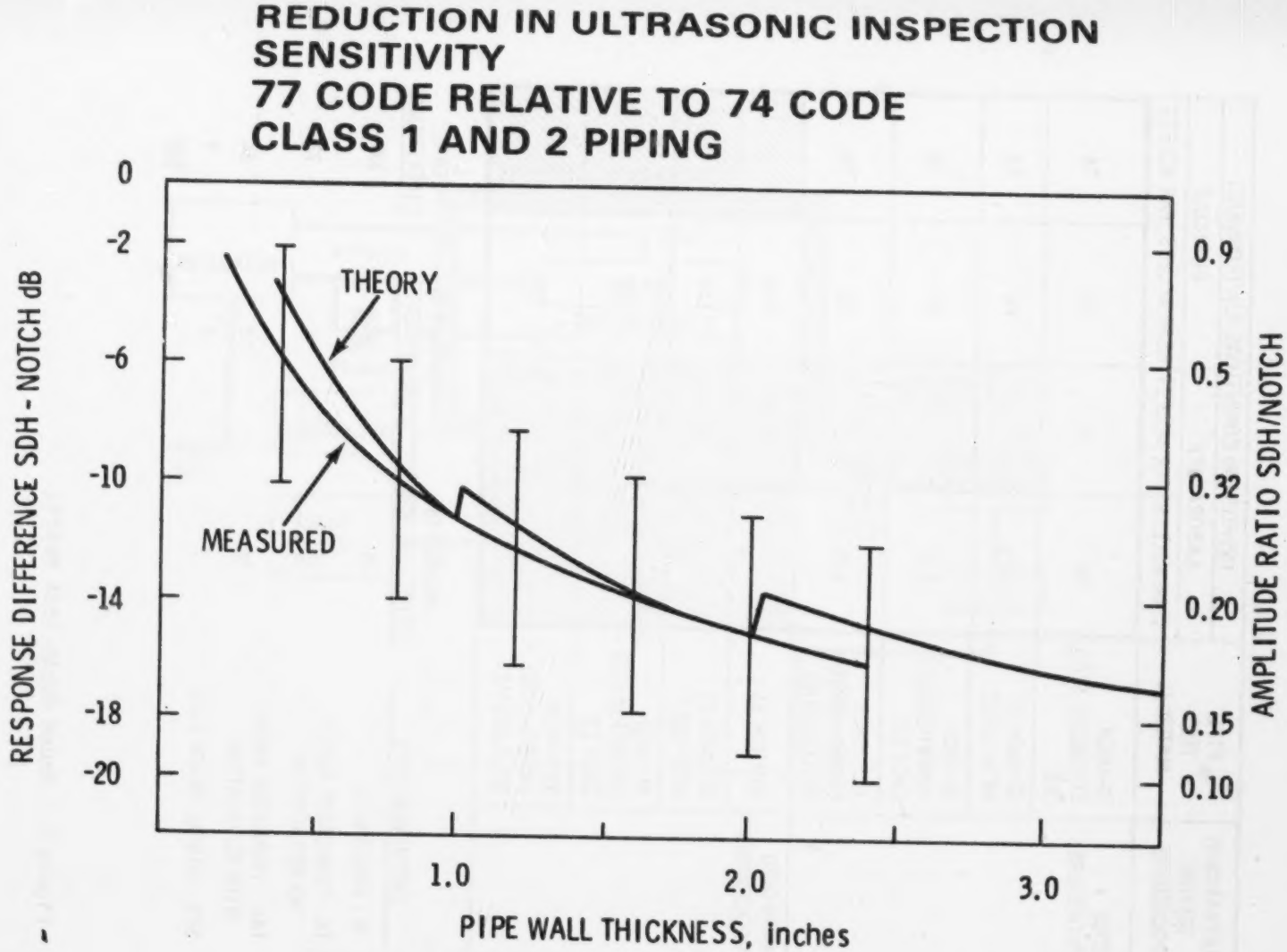


Figure 3. Measured and Theoretical Reduction in Ultrasonic Piping Inspection Sensitivity of the 1977 Edition of the ASME Code, Section XI (Notch), as compared to the 1974 Edition (Side Drilled Holes, SDH).

ULTRASONIC TESTING PROCEDURE	PIPE TEST MATERIAL	INSPECTION CONDITIONS (ENVIRONMENT)			
		LABORATORY		DIFFICULT	
		NEAR ACCESS	FAR ACCESS	NEAR ACCESS	FAR ACCESS
CODE PROCEDURES	10-INCH STAINLESS STEEL (SS)	TA	TA	FA	TA
	12-INCH SS WITH IGSCC	SCS		FA	TA
	32-INCH CENTRIFUGALLY CAST SS	SCS		FA	TA
	33.5 -INCH CARBON STEEL + SS CLADDING	SCS		FA	TA
IMPROVED PROCEDURES	10-INCH SS			TAR	
	12-INCH SS WITH IGSCC			TAR	
	32-INCH CENTRIFUGALLY CAST SS			TAR	
	33.3-INCH CARBON STEEL + SS CLADDING			TAR	

EXPERIMENT TYPE	NUMBER OF INSPECTION TEAMS	NUMBER OF EXPERIMENTS	TOTAL EXPERIMENTS
FA - FULL ANOVA	25	4	100
TA - TRUNCATED ANOVA, NO REPLICATION	12	6	72
TAR - TRUNCATED ANOVA, WITH REPLICATION	16	4	64
SCS - SINGLE CRACK SIZE	3	3	9
			245

Figure 4. Round Robin Test Matrix.

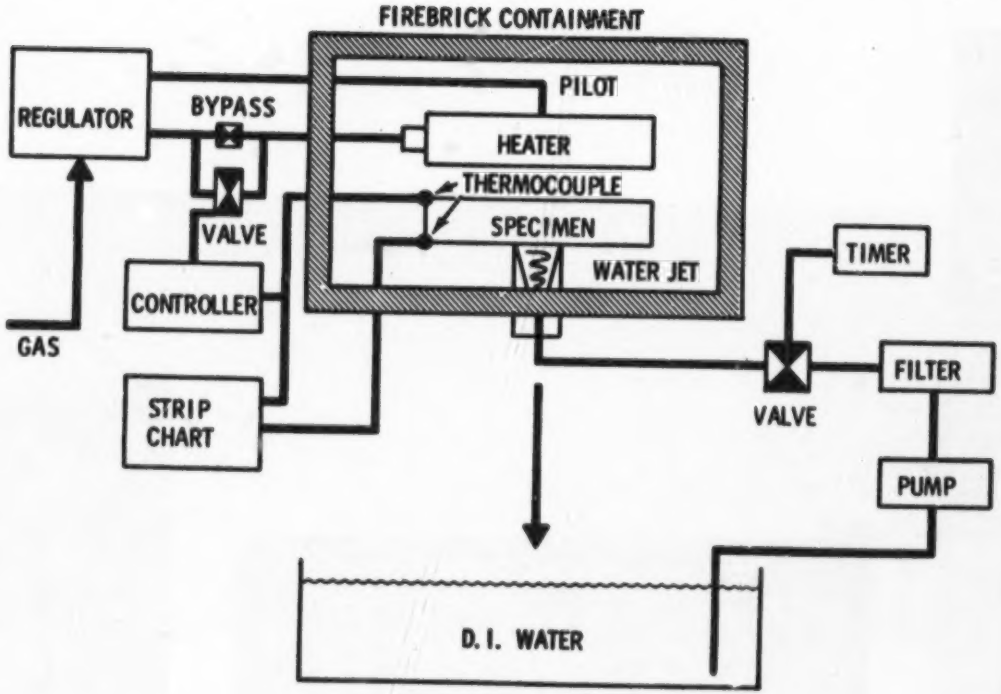


Figure 5. Thermal Fatigue Facility Block Diagram

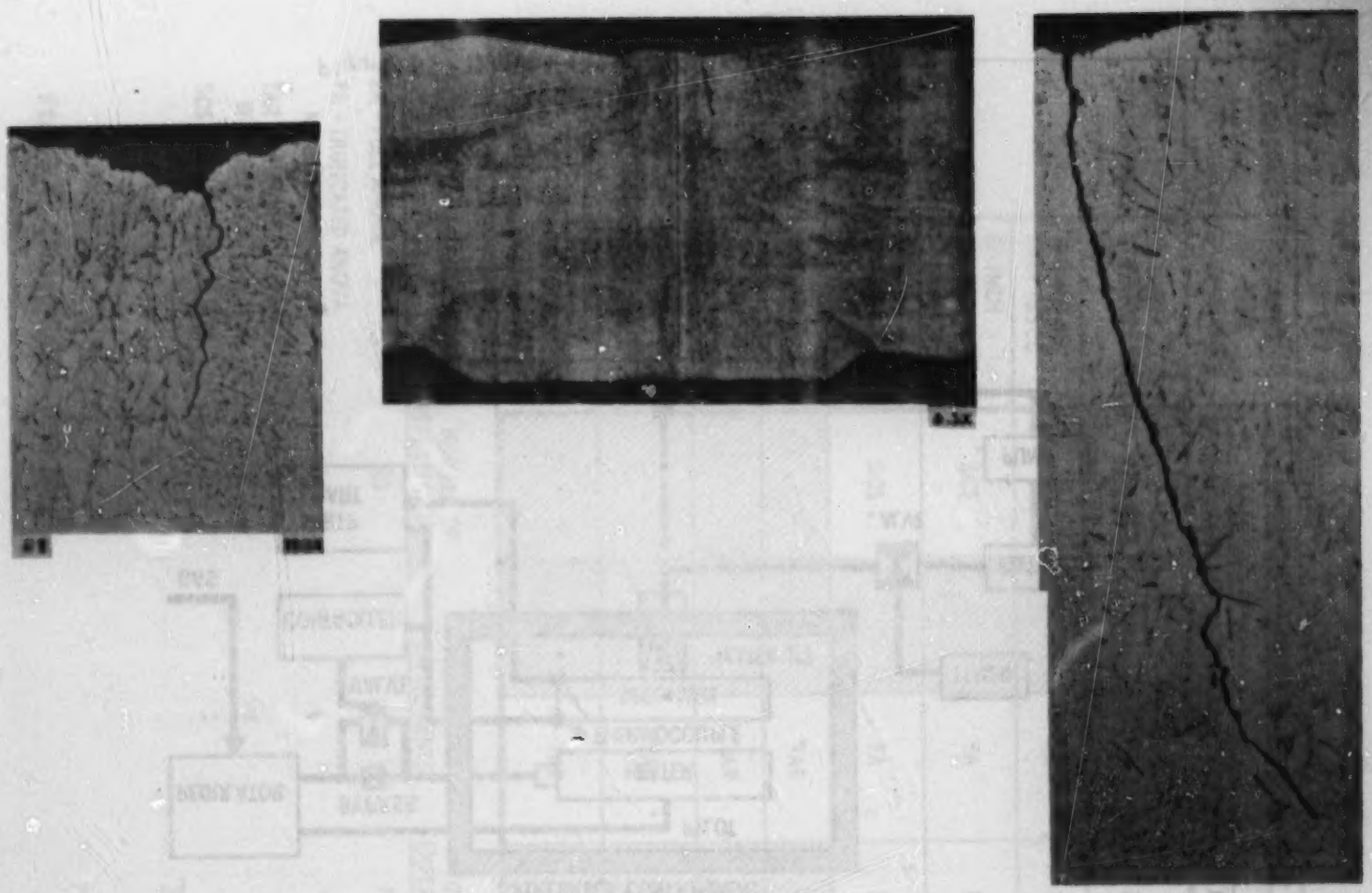


Figure 6. Thermal Fatigue Crack 10 Inch Schedule 80 Stainless Pipe.

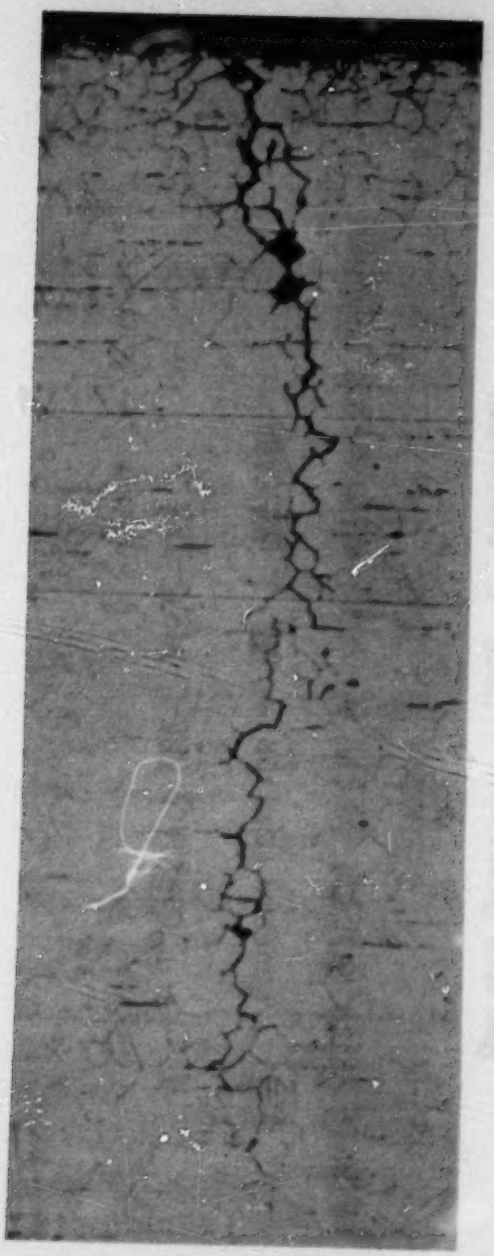


Figure 7. IGSCC Crack in 10 Inch Schedule 80 Stainless Pipe.

REAL-TIME SAFT-UT FOR FLAW IMAGING AND DISPLAY TECHNIQUES

CONTRACT NRC-C-04-75-182

PROGRAM MONITOR -- DR. J. MUSCARA

EIGHTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING
OCTOBER 31, 1980

PROJECT DIRECTOR -- PROF. S. GANAPATHY

DEPARTMENT OF ELECTRICAL AND COMPUTER ENGINEERING
UNIVERSITY OF MICHIGAN
ANN ARBOR, MI 48109

LICENSING AND SAFETY ISSUES IN NDE OF REACTOR VESSELS

- CONVENTIONAL NDE TECHNIQUES NEED IMPROVEMENT IN FLAW SIZING AND CHARACTERIZATION.
- IMPROVEMENTS NEEDED IN ACCURACY AND RELIABILITY.
- FRACTURE MECHANICS CAN (USING MORE ACCURATE RESULTS) DETERMINE SAFETY OF OPERATION OF REACTOR VESSELS.
- MINIMIZE ERRORS IN SUBJECTIVE JUDGEMENT AND DECISION MAKING.
- THESE IN TURN WILL RESULT IN IMPROVED CODES AND LICENSING PROCEDURE.

PROGRAM OBJECTIVES AND SCOPE

OBJECTIVE: TO DESIGN AND IMPLEMENT A REAL-TIME NDE SYSTEM USING SAFT-UT TECHNIQUES FOR FIELD TESTING.

SCOPE:

- IMPROVE THE OPERATIONAL SPEED OF SAFT-UT TECHNIQUE (COMPUTER ARCHITECTURE).
- PROVIDE ENHANCED GRAPHICAL DISPLAY FOR IMPROVED FLAW VISUALIZATION (COMPUTER GRAPHICS).
- DESIGN AND IMPLEMENT TECHNIQUES FOR IMAGING FRONT SURFACE FLAWS (SIGNAL PROCESSING).
- CALIBRATION AND CHARACTERIZATION OF SAFT-UT SYSTEM (ACOUSTICS/OPTICS).
- AUTOMATIC DECISION MAKING REGARDING FLAW-SIZING ORIENTATION, TYPE, ETC., (IMAGE PROCESSING).
- DESIGN EFFICIENT TECHNIQUES FOR FLAW DETECTION (SIGNAL PROCESSING/COMMUNICATION THEORY).
- IDENTIFY BOTTLENECKS IN REAL-TIME SYSTEM AND FIND SOLUTIONS (SYSTEM ANALYSIS).

SUMMARY OF RESULTS

- ORDERS OF MAGNITUDE IMPROVEMENT OBTAINED IN PROCESSING SPEED.
- SEVERAL INTERACTIVE DISPLAY ROUTINES HAVE BEEN DESIGNED AND IMPLEMENTED.
- FRONT-SURFACE FLAW IDENTIFICATION OBTAINED USING TWO DIFFERENT TECHNIQUES.
- QUALITATIVE AND ANALYTICAL STUDY COMPLETED ON CHARACTERIZATION OF SAFT-UT SYSTEM.
- EDGE DETECTION TECHNIQUES TO AUTOMATICALLY EXTRACT FLAW BOUNDARIES HAVE BEEN IMPLEMENTED.
- BOTTLENECKS IN REAL-TIME SAFT-UT SYSTEM HAVE BEEN IDENTIFIED.

WORK IN PROGRESS

- ARRAY PROCESSORS/SPECIAL PURPOSE PROCESSOR FOR REAL-TIME SAFT-UT.
- REAL-TIME GRAPHICS USING WALK-THRU TECHNIQUES.
- DECONVOLUTION APPROACH USING INCREASED DYNAMIC RANGE FOR FRONT-SURFACE FLAWS.
- QUANTITATIVE EVALUATION OF SAFT-UT BASED ON PREVIOUSLY REPORTED QUALITATIVE ANALYSIS.
- MATCHED FILTER TECHNIQUES OPTIMIZED FOR GRAIN NOISE IN STAINLESS STEEL.
- STUDY OF SEVERAL EDGE ENHANCEMENT TECHNIQUES FOR ULTRASONIC IMAGES.
- INVESTIGATION OF POTENTIAL SOLUTIONS FOR REAL-TIME SAFT-UT SYSTEM.

REFERENCES

1. "IMPROVED ULTRASONIC NONDESTRUCTIVE TESTING OF PRESSURE VESSELS" (TECHNICAL REPORT), NUREG-0007-2.

2. "IMPROVED ULTRASONIC NONDESTRUCTIVE TESTING OF PRESSURE VESSELS" (TECHNICAL REPORT), NUREG/CR-0135.

3. "IMPROVED ULTRASONIC NONDESTRUCTIVE TESTING OF PRESSURE VESSELS" (TECHNICAL REPORT), NUREG/CR-0581.

4. "IMPROVED ULTRASONIC NONDESTRUCTIVE TESTING OF PRESSURE VESSELS" (TECHNICAL REPORT), NUREG/CR-0909.

5. J.R. FREDERICK, S. GANAPATHY, ET.AL., "IMPROVED CHARACTERIZATION OF DISCONTINUITIES IN THICK WALLED PRESSURE VESSELS", CONFERENCE ON THE PERIODIC INSPECTION OF PRESSURE VESSELS, LONDON, MAY 1979.

6. S. GANAPATHY, "ULTRASONIC IMAGING TECHNIQUES IN NONDESTRUCTIVE EVALUATION," PROCEEDINGS OF CSNI SPECIALIST MEETING ON RELIABILITY OF ULTRASONIC INSPECTION OF AUTOMATIC STAINLESS STEEL COMPONENTS, BRUSSELS, BELGIUM, MAY, 1980.

INSERVICE INSPECTION APPLICATION OF THE SYNTHETIC APERTURE FOCUSING TECHNIQUE FOR ULTRASONIC TESTING (SAFT UT)

Jerry L. Jackson
Assistant Director
Research and Development

and

Dennis R. Hamlin
Senior Research Engineer
Quality Assurance Systems and Engineering Division
Southwest Research Institute

Presented at

1980 Safety Research Information Meeting

October 27-31, 1980
Gaithersburg, Maryland

Work Sponsored by

Metallurgy and Materials Research Branch
Division of Reactor Safety Research
Nuclear Regulatory Commission
NRC Contract NRC-04-77-145

INSERVICE INSPECTION APPLICATION OF THE SYNTHETIC APERTURE FOCUSING TECHNIQUE FOR ULTRASONIC TESTING (SAFT UT)

Jerry L. Jackson
Assistant Director
Research and Development

and

Dennis R. Hamlin
Senior Research Engineer
Quality Assurance Systems and Engineering Division

BACKGROUND

Southwest Research Institute (SwRI) is conducting a project entitled "Program for Field Validation of the Synthetic Aperture Focusing Technique for Ultrasonic Testing (SAFT UT)." The project's purpose is to continue the development of SAFT UT and to validate its usefulness for inservice inspection applications. SAFT UT is an ultrasonic imaging method for accurately measuring the spatial locations and extent of flaws contained in objects, such as structural components and weldments in nuclear power reactor systems. The increased measurement accuracy offered by SAFT UT, when compared with that provided by measurement methods now in use, will improve the reliability of flaw severity assessments with resultant safety and economic benefits to the nuclear power industry.

This project is one part of a team effort being conducted by the University of Michigan (U of M) and SwRI. Although the two groups are working under separate contracts, their efforts are integrated and synergetic. The U of M group is concentrating on theoretical aspects of SAFT UT while SwRI, working with information and advice supplied by U of M, is contributing the engineering and performance evaluations necessary to produce and validate a practical field tool.

The objectives of this program are to:

- Produce a SAFT UT system in a form useful for field test.
- Validate the performance of that system for characterizing both real and simulated flaws in realistic test specimens.
- Conduct field trials using the system on actual structures.
- Estimate the reliability and confidence level for SAFT UT examinations and compare those results with similar estimates made for conventional ultrasonic methods.

The ultimate goal is to produce a field-rated system for highly accurate flaw size measurements and to conduct sufficient testing of that system to promote its acceptance by industrial and regulatory authorities.

Current work is an extension of an earlier contract in which a SAFT UT system was engineered and fabricated. Due to technical difficulties and resultant delays, that project closed before totally satisfactory operation was obtained. Tests indicated that the system possessed the expected resolution, but that the images had rather low signal-to-noise ratio (SNR) and image clarity suffered excessively. The purpose of the current contract is to diagnose and correct the reasons for the low SNR and to conduct tests and evaluations from which the reliability and confidence level associated with this method can be determined.

PROJECT STATUS

Currently, all efforts are concentrated on diagnosing the cause of low SNR and effecting improvements. To be thorough, every aspect (i.e., electronic subsystems, computer coding, and ultrasonic parameters) of the system is being reviewed and tested independently. The rest of this report will discuss results to date.

Data Acquisition System

The data acquisition system is operational and is being subjected to diagnostic tests designed to detect unsuspected errors. To date, no such discrepancies have been detected. Several modifications have been made to improve the reliability of the data acquisition system, and these changes have been validated. At the present time, it is believed that the data acquisition system is completely operational and capable of good performance.

Reconstruction Algorithm and Computer Coding

The reconstruction algorithm used in the SAFT UT system at SwRI is, in detail, somewhat different than the algorithm used by U of M. To guard against the possibility of an error, the reconstructed algorithm was supplied to U of M for their review. This task has been completed, and the joint opinion of the two groups is that the SwRI algorithm is valid. The computer codes which execute the reconstruction algorithm are presently being reviewed by programmers, and several inconsistencies in the data resulting from reconstruction have been detected. Some of these discrepancies appear to be highly systematic while one is very subtle with no recognized pattern. Programmers are busy tracking down the cause for these errors and will make the necessary corrections.

Work at U of M has resulted in perfecting very effective methods for smoothing and displaying SAFT UT images. They have shown the value of applying signal processing methods such as deconvolution and envelope detection for the purpose of removing nonrelevant features from reconstructed images. They have also demonstrated how image clarity and interpretability are improved by displaying SAFT UT images on a color graphic display system. These methods, now being incorporated into the SAFT UT system at SwRI, will improve the clarity of images produced by its system.

Transducer Evaluations

SAFT UT systems in use at U of M and at SwRI use focused transducers mounted one focal length from the component being examined to produce a point-like source of acoustic energy located on the component's surface. The systems were

designed using the theory that acoustic energy propagates into the component as an expanding hemispherical wave front. When this wave front intercepts a small reflector, the scattered energy will propagate in the form of a second expanding spherical wave front whose center is the reflector's location. Because energy is transported within the component by two separate spherically expanding wave fronts, the magnitude of reflected signals diminishes as the inverse fourth power of distance between the quasi point source on the surface and the reflector. As a result, it must be expected that signals detected by a SAFT UT system will be much weaker than signals detected by a conventional system. To overcome this problem, it is important to select construction and experimental operation parameters for the focused transducer which will optimize the SNR of SAFT UT images.

As a part of the overall effort to improve the SNR of images produced by its SAFT UT system, SwRI has initiated comprehensive experiments to quantify the effect of construction and experimental operation parameters for focused transducers. These experiments are designed to answer questions such as:

- How does the numerical aperture of the focused transducer affect the sensitivity of the transducer and the directivity pattern of the quasi point source.

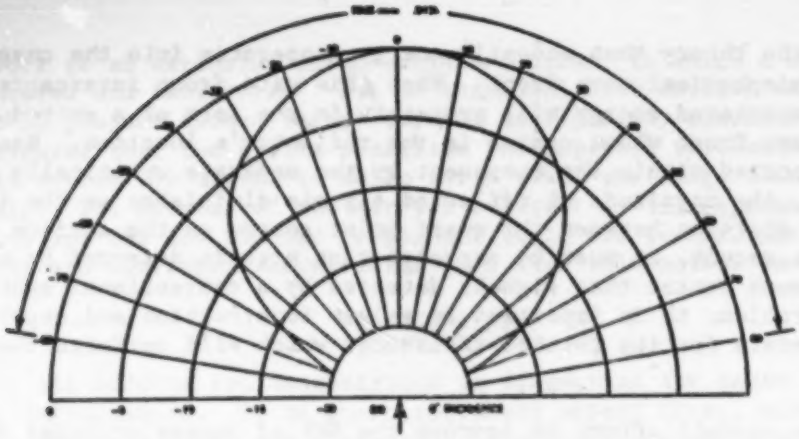
- How valid is the assumption that transmitted energy propagates as an expanding hemisphere. The success of present reconstruction algorithms is directly dependent on the answer to this question.

- How do changes of incident angle affect the directivity pattern of the quasi point source.

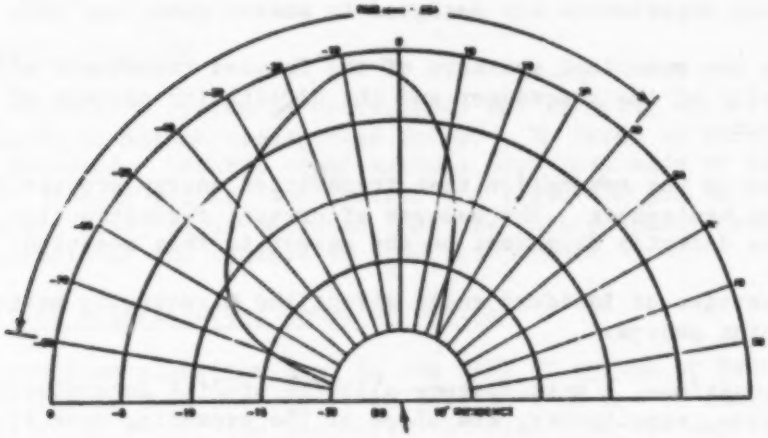
To answer these questions, a test fixture allowing precise determination of directivity patterns, sensitivity, and shape of the expanding wave front has been constructed and recently put into service. Characterization of a number of focused transducers having a wide spread of parameters has begun; and, when completed, these results will be collated and tabulated into a set of reference guidelines to be used in the selection of appropriate transducer parameters for widely differing examination conditions and goals.

Although this work has just begun, several important findings have already been ascertained. First, it has been verified by modeling that the propagation of acoustic energy as an expanding sphere is very accurate over a broad angular region. Furthermore, it has been determined that, within the angular region for which the spherical model is valid, the transit time of energy into the component is unaffected by changes of incident angle. Two examples of these measurements are illustrated in Figure 1. These examples show the angular region for which the spherical model is valid as a function of three incident angles--0, 10, and 20 degrees (referenced from normal). Within the angular regions shown, the maximum deviation from a spherical wave front was less than 0.2 wavelength; and the root-mean-square error over the complete angular region was always less than 0.1 wavelength. These results are very satisfactory and validate the design basis for existing reconstruction algorithms.

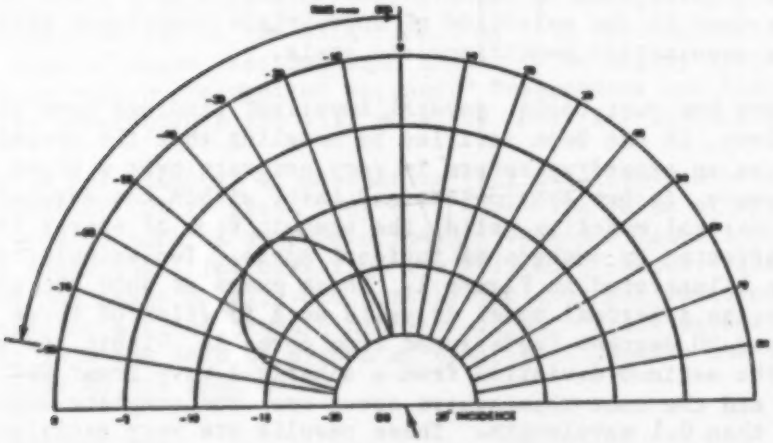
This work is continuing; and, when complete, results will be presented in one of the program progress reports.



(a)

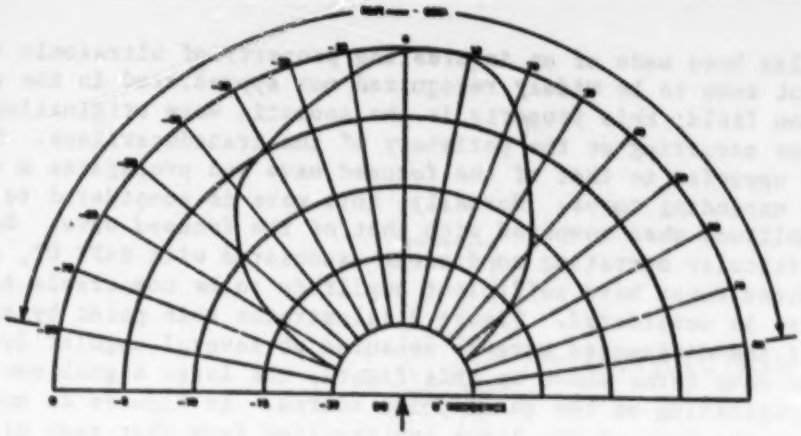


(b)

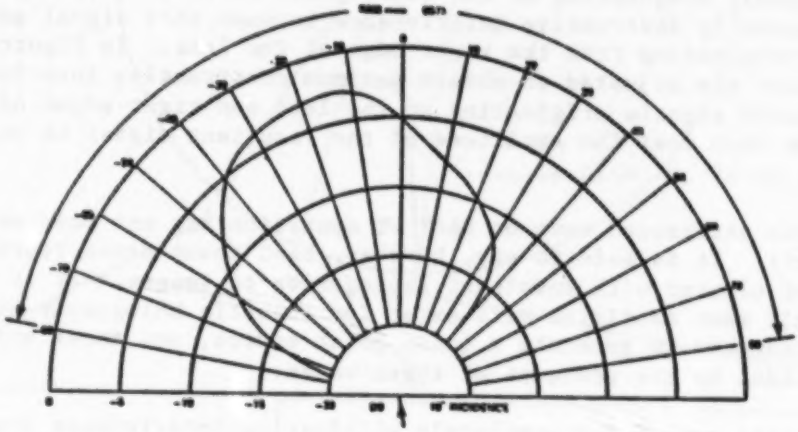


(c)

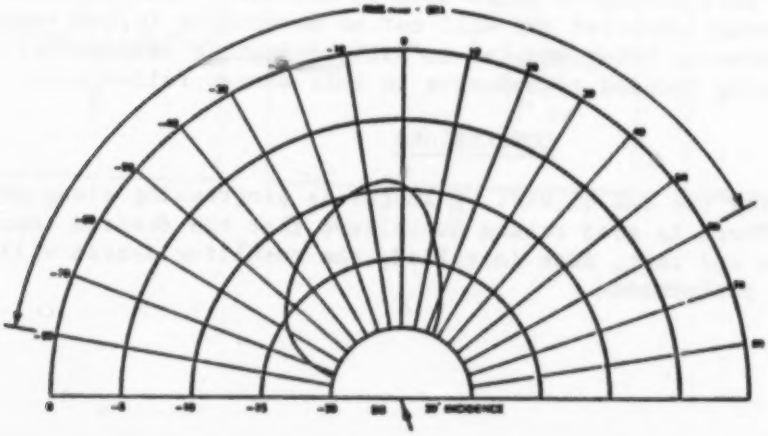
FIGURE 1. MEASUREMENTS OF ACOUSTIC FIELD FOR TWO FOCUSED TRANSDUCERS USED TO APPROXIMATE A POINT SOURCE. FIGURES (a), (b), AND (c) DESCRIBE THE FIELD PRODUCED BY A 3/4-INCH DIAMETER, 1-1/2-INCH LENS POSITIONED TO 0°, 10°, AND 20° INCIDENCE.



(d)



(e)



(f)

FIGURE 1 (Cont'd). FIGURES (d), (e), AND (f) SHOW SIMILAR RESULTS FOR A 3/4-INCH DIAMETER, 1-INCH LENS. NOTE THE BROAD ANGULAR RANGE OVER WHICH THE WAVEFRONT CLOSELY APPROXIMATES A SPHERE.

Measurements have also been made of an interesting property of ultrasonic transducers which does not seem to be widely recognized nor appreciated in the nondestructive examination field; this property is the acoustic wave originating by diffraction processes occurring at the periphery of the transducer/lens. This wave has a polarity opposite to that of the focused wave and propagates a wave from the lens as an expanding torus. Normally, this wave is considered to be insignificant in amplitude when compared with that of the focused wave. However, due to the particular operating conditions associated with SAFT UT, such is not the case. These waves have sufficient amplitude to be comparable to the desired wave and must be considered. Figure 2 illustrates this point by showing the amplitude of the diffracted wave as measured at several angular locations. In all three wave forms shown in this figure, the large signal was produced by the wave originating at the quasi point source. In Figures 2a and 2b, the smaller signal to the left of the large one resulted from that part of the diffracted wave which originated at the left edge of the lens. Also, Figure 2b shows the large signal, originating at the quasi point source, to be asymmetrical. This is caused by destructive interference between that signal and the diffracted signal originating from the right edge of the lens. In Figure 2c, the measurement point was adjusted to obtain maximum constructive interference between the diffracted signals originating at the left and right edges of the lens; and it can be seen that the amplitude of the resultant signal is only a few dB less than that of the main signal.

The influence of the diffracted wave on SAFT UT operation has not been determined quantitatively. It is safe to say, however, that these waves represent a potential source of clutter with resultant degradation to images. It is worth mentioning that this same condition will exist for acoustic holography systems using focused transducers to generate a quasi point source, and those systems could also be degraded by the presence of these waves.

A simple and practical method for completely eliminating interference from these waves is to mount an absorbing mask near the surface of the component and to pass the focused wave through a small hole in the mask. The diffracted waves will be completely absorbed and will not be detectable in the examination signals. This practice is being adopted at SwRI, and it is recommended that other researchers using focused transducers in this manner follow suit.

CONCLUSIONS

Work towards improving the SNR of SAFT UT images is progressing along several concurrent paths. There is good reason to believe that the desired improvements are obtainable and that, once installed, the resulting system will exhibit the desired performance.

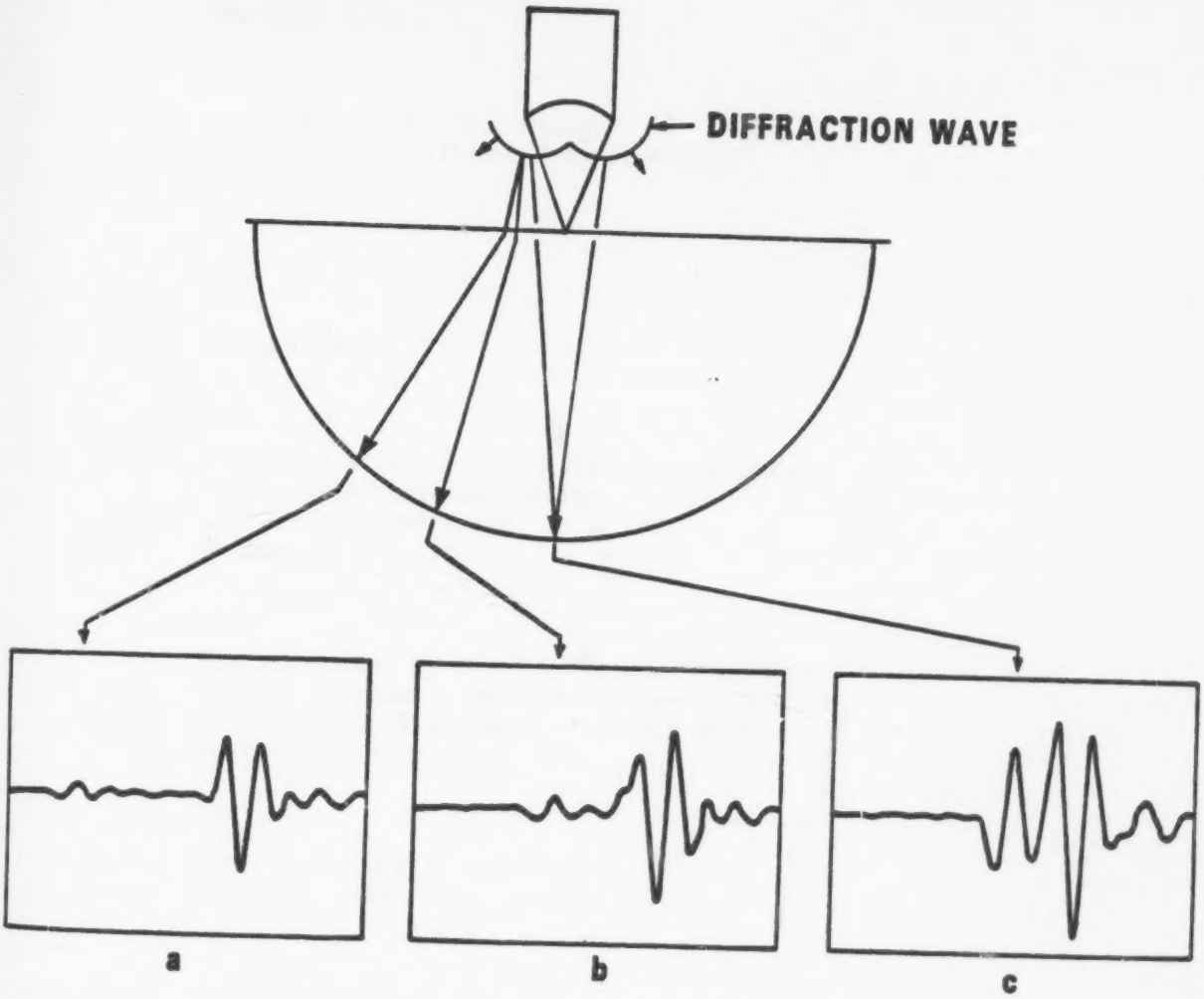


FIGURE 2. SIGNALS FROM AN ACOUSTIC WAVE PRODUCED BY EDGE DIFFRACTION. IN (c), NOTE THE LARGE RESPONSE CAUSED BY CONSTRUCTIVE INTERFERENCE OF WAVES EMITTED AT LEFT AND RIGHT EDGES OF THE LENS. SURROUNDING THE FOCUSED WAVE BY AN ABSORBING MASK ELIMINATES THESE UNDESIRED SIGNALS.

ACOUSTIC EMISSION - FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS

P. H. Hutton, Project Manager
R. J. Kurtz, Assistant Project Manager

October 1980

Presented at the Eighth Water Reactor Safety Research Information Meeting
Gaithersburg, Maryland
October 27-31, 1980

Work Supported by the U.S. Nuclear Regulatory Commission under a Related Services Agreement with the U.S. Department of Energy under Contract DE-AC06-76RLO 1830

Pacific Northwest Laboratory
Richland, Washington 99352

ACOUSTIC EMISSION - FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS[a]

Contractor: Pacific Northwest Laboratories
P.H. Hutton, Project Manager
R.J. Kurtz, Assistant Project Manager

Sponsor: NRC, RSR, Materials & Metallurgy Research Branch
Dr. J. Muscara, Program Manager

OBJECTIVE

The objective of the acoustic emission (AE)/flaw characterization program is to provide an experimental feasibility evaluation of using the AE method on a continuous basis (during operation and during hydrotest) to detect and analyze flaw growth in reactor pressure vessels. This effort is based on the philosophy that AE offers the potential of being a valuable addition to current NDI methods with unique capability for continuous monitoring, high sensitivity, and remote flaw location.

LICENSING AND SAFETY ISSUE

This program addresses the following areas of significance:

- Older reactors where effective inspection of the vessel by conventional methods is extremely difficult. AE can potentially be used to monitor these vessels to detect and locate active flaws, facilitate an estimate of severity based on AE, and localize shielding penetration location(s) for flaw inspection by conventional methods.

- Monitor vessel areas such as nozzles where conventional NDI is difficult and expensive. AE could detect the presence of an active flaw and maintain surveillance of flaw growth to minimize the need for conventional NDI.

- As a secondary benefit, AE systems provide a sensitive detector of leaks as well as cracking in piping. They can also be adapted to sensing flow - no flow in critical valves.

---

[a] Prepared for the 8th Water Reactor Safety Research Information Meeting, National Bureau of Standards, Gaithersburg, MD, October 27-31, 1980.

SCOPE

The program scope is described by three primary areas of effort:

- Develop a method to identify crack growth AE signals as unique from other innocuous but similar acoustic signals.

- Develop a relationship between measured AE and crack growth which will enable an estimate of flaw severity based on measured AE information.

- Demonstrate the total concept through off-reactor vessel tests and finally, on-reactor monitoring. This includes developing the necessary instrumentation system.

The program is structured to start with testing laboratory specimens to determine fundamental feasibility of program objectives using ASTM A533B, Class 1 steel. Since theoretical transfer of these results to a full size structure is very questionable in this case, the next phase calls for testing on a heavy section (> 4 inch wall) vessel to check laboratory results and establish criteria more directly relatable to a reactor vessel. Vessel testing is to include a simulation of pertinent reactor environment conditions (background noise, flaws exposed to pressurized and heated water, etc.) excluding nuclear radiation. The final phase requires installation and operation of a prototype AE monitor system on an operating reactor on a test basis to check vessel results and finalize application methods and system.

RESULTS

Major accomplishments to date include:

- Completion of laboratory testing from which we:

  (a) showed the feasibility of separating crack growth AE signals from other transient signals using pattern recognition methods

(b) developed AE/fracture mechanics relationships for flaw severity interpretation for fatigue crack growth and hydrotest loading.

- Established a location and facilities for performing simulated reactor vessel monitoring.

- An AE monitor system to implement laboratory results has been designed and is about 75% fabricated.

- Negotiating for installation of an AE sensing system on a reactor.

Expanding on the accomplishments:

Vessel Test

The purpose of this test is to determine the validity of laboratory developed application methodologies when applied to AE data obtained from fatigue crack growth and fracture in a heavy section steel pressure vessel tested under conditions which attempt to simulate reactor pressure vessel operation. Specifically, the principal objective is to evaluate AE signal identification and flaw severity assessment techniques which have been developed from testing laboratory specimens of A533B, Class 1 steel.

Much of the effort this past year has concentrated on locating an appropriate site for conducting this test and developing a test plan which fulfills the principal objective. The Materialprüfungsanstalt (MPA) Stuttgart laboratory plans to perform a test of an intermediate scale pressure vessel, beginning in early CY-1981. PNL has been invited to participate in this test under the auspices of the NRC-BMFT Safety Research Exchange Agreement.

The MPA test vessel, ZB1, is a 4.7 inch thick, 58 inch I.D. cylindrical vessel made from nominally A508 steel (see Figures 1 and 2). Since this program is focused on A533B steel, the current plan is for PNL to supply

an insert of A533B steel with machined and fatigue presharpened flaws to be welded into ZB1 (see Figures 2 and 3). The insert will contain three part-circular surface flaws; two on the inside surface and one on the outside surface. The longitudinal axes of the flaws shall be perpendicular to the circumferential stress in the vessel.

The test plan is designed to evaluate flaw detection and evaluation by AE methods under both operating and hydrotest conditions. In order to simulate reactor pressure vessel loading conditions, the test vessel will be internally pressurized with water at temperatures of 200 and 550°F. The water pressurization system will provide constant amplitude cyclic, as well as monotonic mechanical stressing of the vessel. The cyclic frequency will be about 1 cycle per minute. Blocks of pressure cycles will be applied at R - ratios (R = minimum pressure/maximum pressure) of 0.1 and 0.6. These high and low R - ratio loading blocks will be alternated, and are designed to produce approximately equal amounts of crack growth at each R - ratio. Hydrotests will also be performed periodically during the test sequence. The first vessel loading will be a simulated hydrotest. Additional hydrotests, identical to the first, will be performed following each R = 0.6 loading block. The final hydrotest, to be conducted after all fatigue cycling is completed, will pressurize the vessel to failure.

One of the important objectives of AE monitoring the vessel test is to evaluate the effect of various reactor background noises on AE monitoring. This relates to both AE detection and crack growth AE identification in the presence of noise signals. We plan to simulate several sources of noise. These include: (1) hydraulic flow noise, (2) mechanical rubbing of thermal insulation, (3) mechanical stressing of a slag inclusion in a fabrication weld, (4) thermal stressing of a base metal-to-weld cladding interface, and (5) electrical transients.

AE Monitor System

The monitor system concept, as shown in Figure 4, utilizes two arrays for the MPA vessel test. One will concentrate on the A533B insert

to maximize AE data acquisition from that area. The second array will monitor the total cylindrical section of the vessel to better simulate in-service monitoring. An upgraded NRC-owned Dunegan/Endevco 1032 AE monitor will be used to determine AE source location and associated signal parameters such as peak time, duration, etc. This information together with the parametric features will be stored on floppy disk and it will also go to the Digital Equipment Corporation PDP 11/03 computer for on-site analysis.

Digitized signal waveforms will be recorded on tape in parallel with processing through the D/E 1032 system. These waveforms will then be played back to the PDP 11/03 for evaluation by pattern recognition.

The PDP 11/03 will be used also to sort pattern recognition information on the basis of a source location derived from the D/E 1032 system. A common waveform numbering will be used to properly correlate pattern recognition results with source location.

Data identified by pattern recognition as crack growth AE and associated with a specific location will be used to estimate flaw severity.

As a contingency backup for the vessel test, a zone isolation subsystem is also provided. In the event that the number of noise signals is so great that the waveform recorder is essentially totally occupied recording these at the sacrifice of AE signals, the zone isolator can be used to restrict waveform acceptance to that from a specified area. This will also serve to identify AE from individual flaws in the A533B insert if the source location system does not provide adequate source resolution.

An important feature of the system is provision for storing all of the basic data (source location, signal features and digitized waveforms). This will facilitate post test analysis to improve and refine AE signal identification and interpretation models.

AE Signal Identification

A pattern recognition algorithm will be applied in the vessel test to distinguish flaw growth AE from other transient acoustic signals. Analysis of composite laboratory data from several sensor-specimen combinations demonstrated the need to calibrate the algorithm on a given sensing system. Since waveguides will be used in the vessel test, the decision function will be based on available waveguide laboratory data wherein a 99% correct classification of AE and noise was achieved.

In Figure 5, a flow diagram shows the implementation of pattern recognition planned for the vessel test. The digitized waveforms will be acquired from the nine-track tape records developed by the waveform recorder (refer to AE Monitor System discussion). The PDP 11/03 computer will be used to perform the determinations shown. Features computed are those effectively applied to laboratory data. In the step combining AE data with source location data, the signals identified as crack growth AE by pattern recognition will be grouped by a source area. These results will then be applied in the flaw severity evaluation model.

Taped waveform data will also be shipped to PNL for analysis with the objective of improving and refining the decision rule.

Flaw Severity Evaluation

Application of AE to flaw assessment is based upon linear elastic fracture mechanics using Section XI, Appendix A, as a foundation for developing the method of application.

To relate AE obtained during reactor operation or hydrotest to fracture mechanics variables requires that experimental data exist for the relevant material and environmental conditions anticipated during pressure vessel service. The objective of the laboratory test phase of this program was to generate AE data from fracture mechanics specimens tested under conditions relevant to reactor operation. The investigation was by design limited to ASTM Type A533, Grade B, Class 1 steel tested in

fatigue crack growth and fracture loading conditions. Testing concentrated on characterizing the effect of base and weld metal, flaw geometry, temperature, and mechanical loading upon the AE response. Concepts for quantitative flaw evaluation during both operation and hydrotest conditions have been derived from the test data.

In Figure 6, a composite plot of the AE/fatigue crack growth data measured from laboratory specimens is presented. The two parallel lines represent conservative lower boundary lines, which approximate the slopes of the room temperature and 550°F test data. Note that the lab data indicates that the AE event count rate (counts/second) depends primarily on the crack growth rate (micro-inches/second) and temperature. Figure 7 shows a procedure for using the experimental data base for estimating flaw severity using AE measured on a reactor. AE events which have similar source location coordinates and have been identified as originating from a crack by pattern recognition analysis are used to construct a curve of event counts versus time. The rate of change of AE events per unit time may then be used with the experimental data to estimate a crack growth rate based on time. This crack growth rate when divided by an estimate of the cycle rate may be used to obtain a measure of the stress intensity factor range (ΔK_I). Since estimating cycle rate is inexact, calculations have been performed using the equations shown in Figure 6 and the da/dn - ΔK equations from Section XI which show that the estimate of ΔK_I is not linearly dependent on cycle rate. An order of magnitude change in the cycle rate changes ΔK_I by less than a factor of two.

AE data obtained from surface notch flaws in 6 inch wall HSST vessel hydrotests is shown in Figure 8 in terms of AE event count versus stress intensity factor, K_I. A concept for applying these results to estimate flaw severity from AE during hydrotest is depicted in Figure 9. During hydrotest, only that quantity of AE data which occurs during the overpressure (i.e., between the nominal reactor operating pressure, P_o, and the maximum hydrotest pressure, P_h) is used in the analysis. Thus for a particular quantity of AE (δN) from a flaw at a given location, and assuming that the mathematical relationship between

AE and K_I may be approximated as a linear function, flaw severity may be estimated from:

$$K_I = \frac{\delta N}{C_2} \frac{P_h}{P_h - P_o}$$

where C_2 is the slope of the nearly linear relationship between AE and K_I.

These relationships will be applied to on-site analysis of data from the MPA vessel test described earlier. Their accuracy depends on:

- Separation of AE from noise signals.
- Consistent response of the AE monitor system.
- The accuracy of experimental relationships between flaw severity and AE for the pressure vessel material under consideration including flaw-to-sensor distance effects and geometric effects.
- The flaw dimensions remaining essentially constant during a hydrotest and the K_I descriptor being valid up to the hydrotest pressure as well as during operation.

The vessel test plan provides for preserving the basic data to facilitate post test analysis to strengthen the relationships where needed prior to testing on-reactor.

Reactor Installation

Based on discussions with TVA personnel at Knoxville, Tennessee, it appears to be technically and administratively feasible to install an AE monitor system on a TVA reactor for test and evaluation. Installation of sensors, preamplifiers and signal leads out of containment would probably be accomplished by January 1981, even though full test monitoring would not start until early 1982 due to reactor completion schedule.

SUMMARY

The initial laboratory test phase of the program is completed and significant progress has been made toward the off-reactor and on-reactor evaluation phases. The program structure is such that each succeeding phase serves to verify and adjust results from the previous phase. Delays in preparations for the MPA vessel test (both in preparation of the A533B insert and in preparation of the vessel in Germany) plus the completion schedule for the available reactor for on-reactor testing make completion of the program in FY-83 as scheduled questionable. At this point, however, our objective is still to complete the program in FY-83.

PUBLICATIONS

Quarterly Report - July 1-September 30, 1979, P.H. Hutton, E.B. Schwenk, R.J. Kurtz, NUREG/CR-1009, PNL-3040-3, March 1980.

Quarterly Report - October 1-December 31, 1979, P.H. Hutton, R.J. Kurtz, NUREG/CR-1349, PNL-3040-4, August 1980.

Quarterly Report - January 1-March 31, 1980, P.H. Hutton, R.J. Kurtz, NUREG/CR-1454, PNL-3380-1, August 1980.

Quarterly Report - April 1-June 30, 1980, P.H. Hutton, R.J. Kurtz, et al., NUREG/CR-1762, PNL-3380-2, November 1980 (anticipated).

Quarterly Report - July 1-September 30, 1980, P.H. Hutton, R.J. Kurtz, et al., NUREG/CR-?, PNL-3380-3, December 1980 (anticipated).

Leak Detection and Crack Detection in Light Water Nuclear Reactor Systems by Acoustic Emission Methods, A Survey Report, P.H. Hutton, submitted to NRC only, March 1980.

Development of Techniques for Continuous Acoustic Emission Monitoring of Light Water Reactors, P.H. Hutton, R.J. Kurtz, paper PNL-SA-8312, ASNT Spring Conference, Philadelphia, PA, March 1980.

CREDITS

Primary among other Battelle staff members contributing to this work are: J.R. Skorpik, T.T. Taylor, J.F. Dawson, E.B. Schwenk, R.T. Landsiedel, G.D. Shearer, P.G. Doctor and T.P. Harrington. Their contributions are gratefully acknowledged.

READEX FICHE best copy available

pages tightly bound

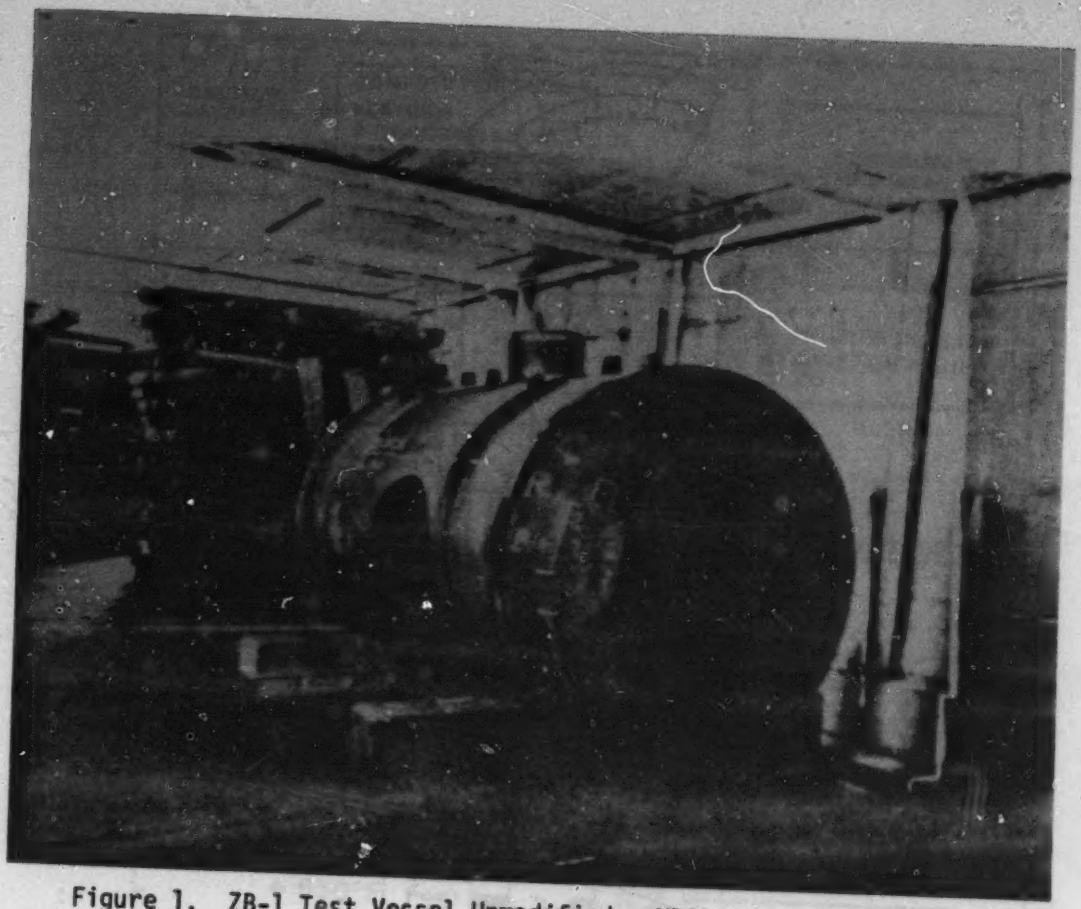


Figure 1. ZB-1 Test Vessel Unmodified - MPA

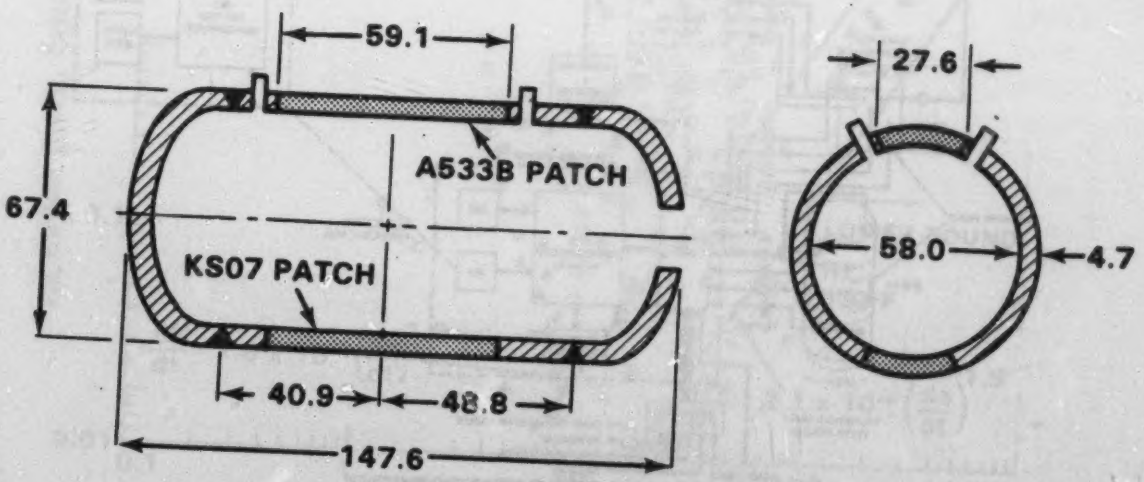


Figure 2. A533B Insert for ZB-1 Vessel

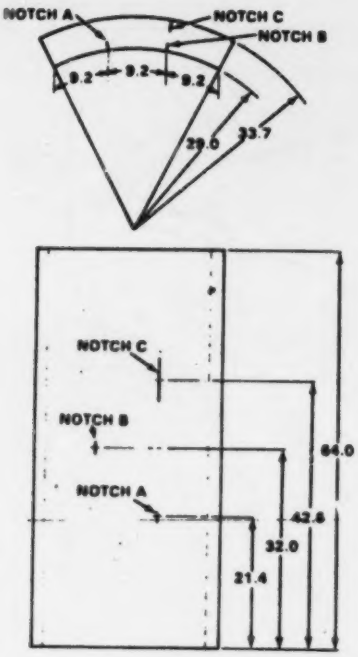


Figure 3. A533B Insert for ZB-1 Vessel

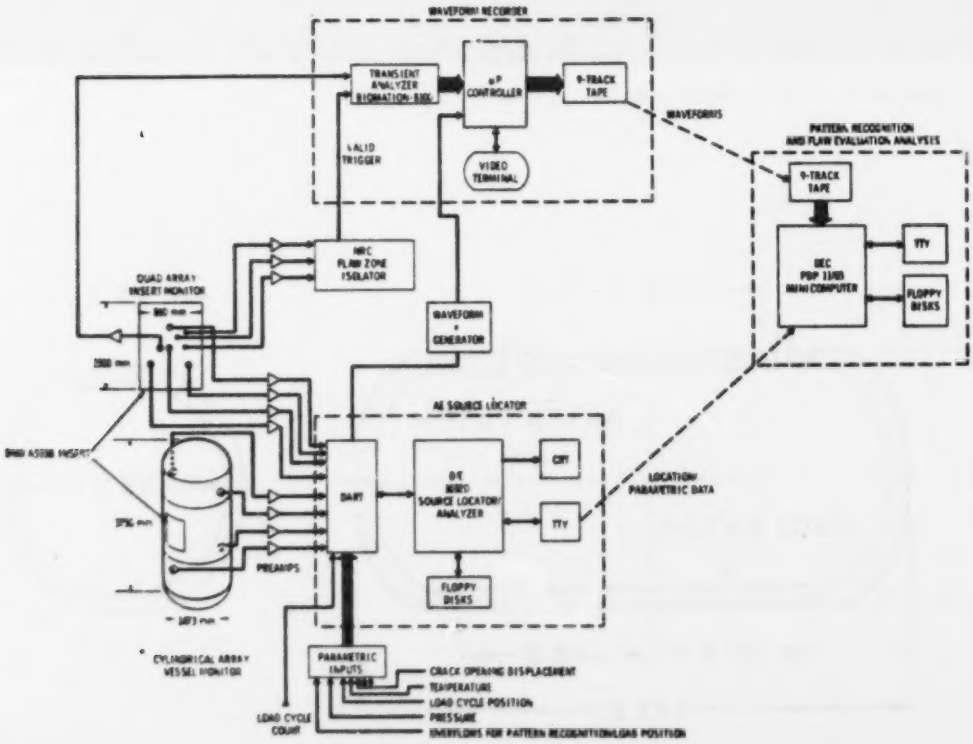


Figure 4. AE Monitor System Concept

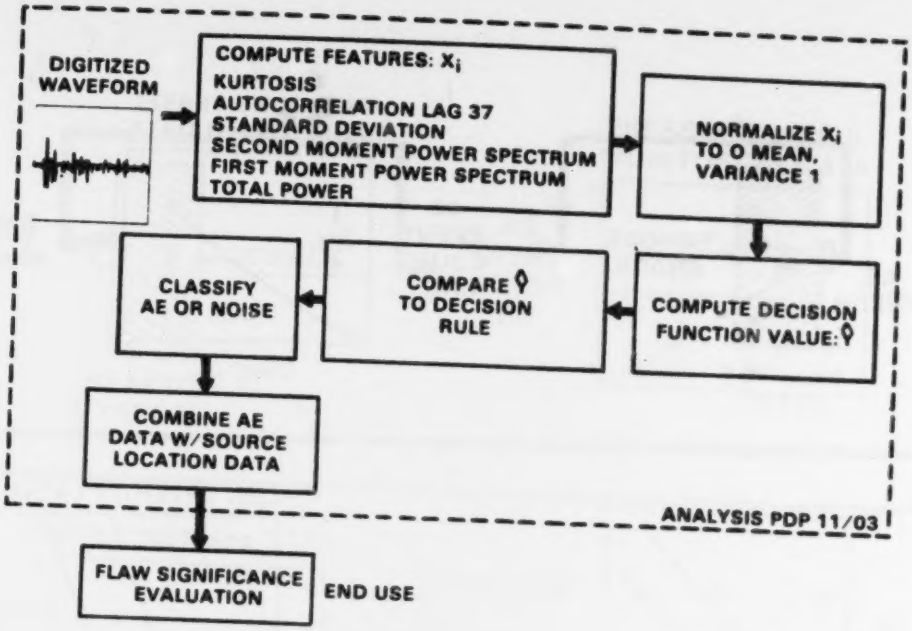


Figure 5. Pattern Recognition Implementation

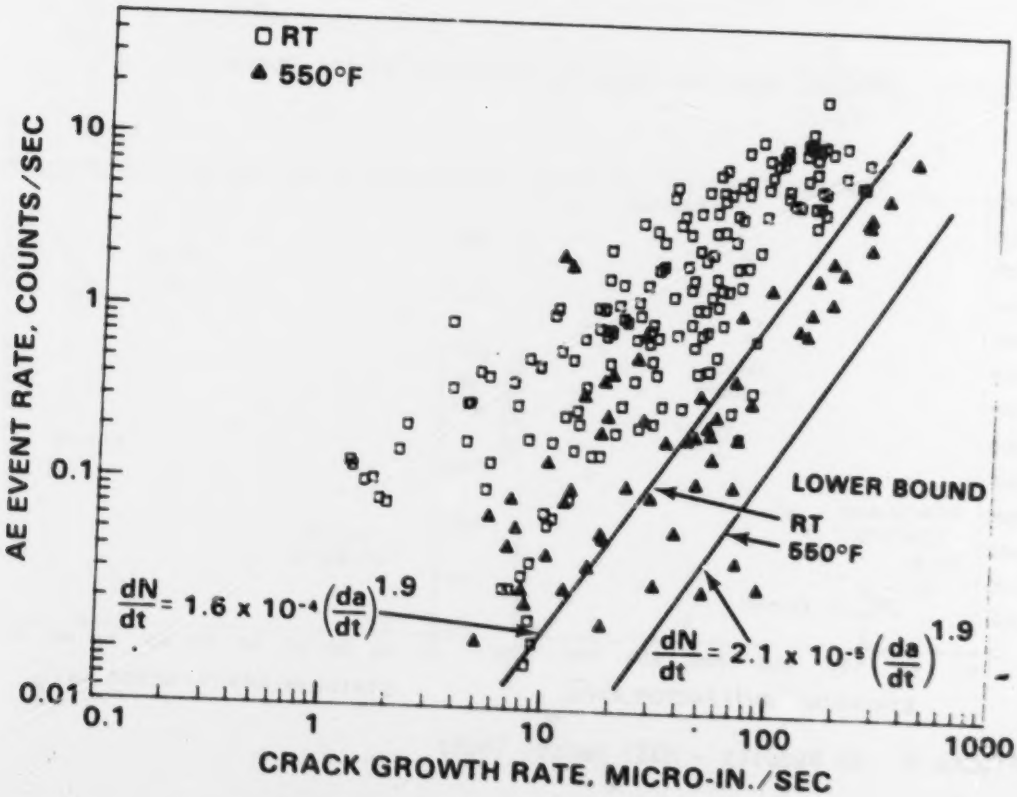


Figure 6. Experimental Fatigue Crack Growth/AE Data

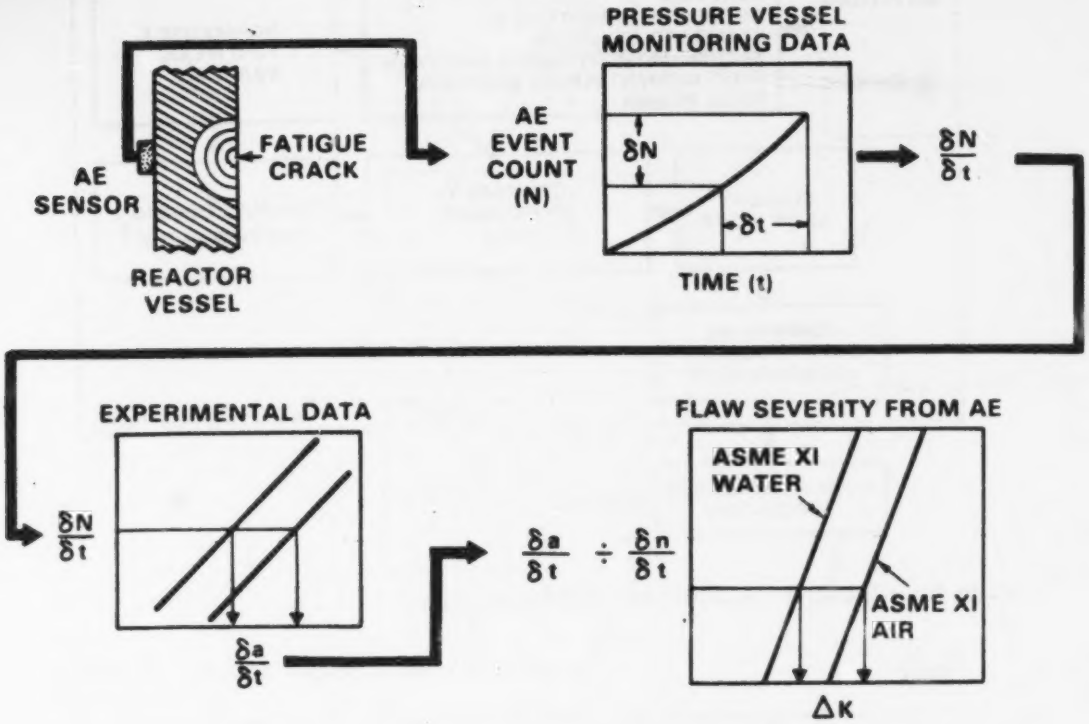


Figure 7. Method for Flaw Severity Estimate During Operation

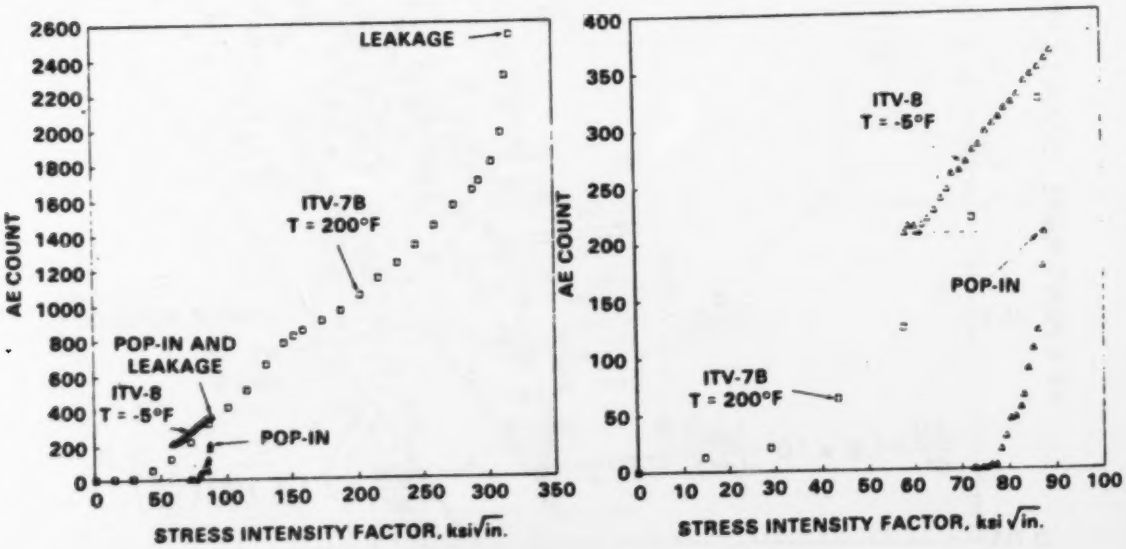


Figure 8. AE Results - HSST Vessel Tests

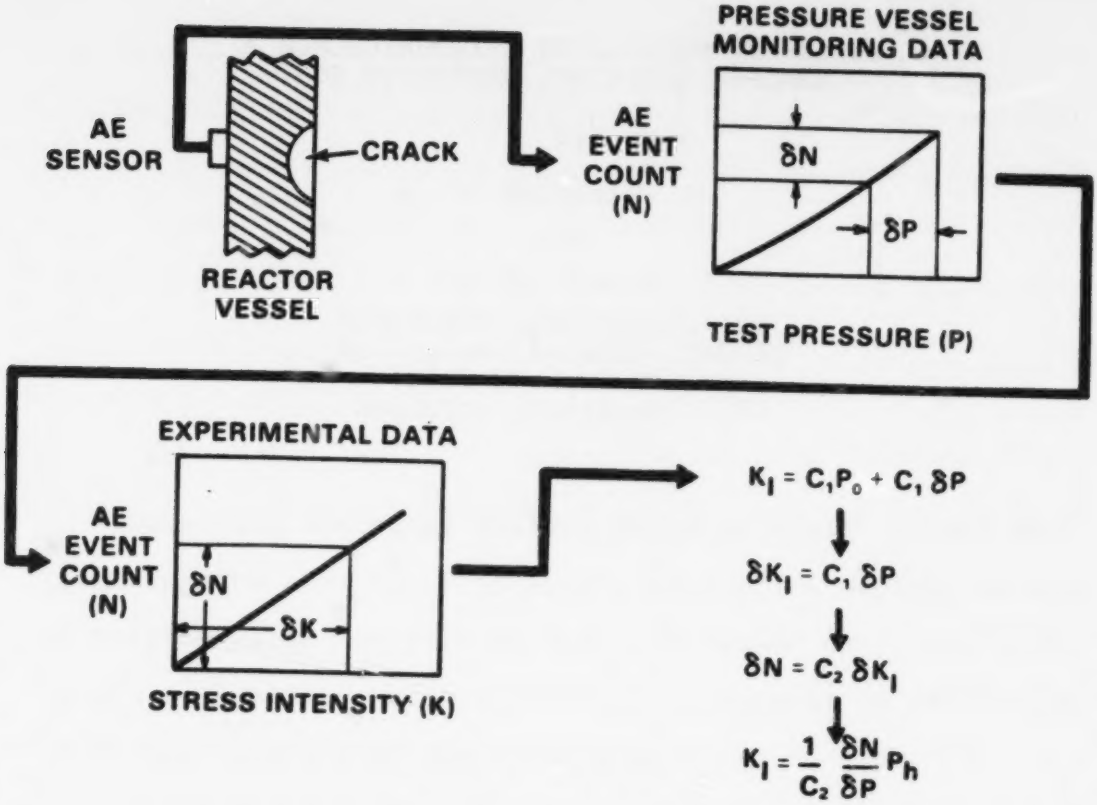


Figure 9. Method for Flaw Severity Estimate During Hydrotest

A NONDESTRUCTIVE EVALUATION TECHNIQUE FOR DETECTING INCIPIENT FAILURE

by

A. A. Hochrein, Jr.

and

L. L. Yeager

DAEDALEAN ASSOCIATES, Inc.
Springlake Research Center
15110 Frederick Road
Woodbine, Maryland 21797

Detecting incipient failure modes in materials and determining the structural integrity of complex piping systems of boiling water reactors in order to assure component safety and reliability have been the major impetus in developing nondestructive evaluation techniques.

DAEDALEAN engineers have developed a unique nondestructive evaluation technique which utilizes the phenomenon of internal friction in materials and identifies changes in the specific damping capacity as a function of the integrity of the material being tested. The program was initiated by the United States Nuclear Regulatory Commission under the auspices of the Reactor Safety Research Program with Dr. Paul Wu as Program Manager.

Specifically, four inch stainless steel pipes simulating bypass lines in boiling water reactors (BWR) have been tested utilizing the internal friction damping nondestructive evaluation (IFD-NDE) technique. Various conditional operating

parameters such as temperature, pressure load range, and oxygen content were included as a part of the overall test design. The analysis of data derived from numerous tests conducted in the laboratory indicate that the different failure mechanisms each have their own characteristic decay associated with failure in the stainless steel pipe. Further, the internal friction measurement which is identified as the specific damping capacity of the various pipes increases as the test pipes progress toward ultimate failure. The identification of the failure mode was associated with a characteristic internal friction measurement and enabled the engineers to immediately assess the integrity of the sample pipe section.

The added safety associated with the IFD-NDE technique would be realized as degraded stainless steel pipes are tested efficiently.

Present nondestructive test methods are useful in determining serious flaws in piping systems. The IFD-NDE technique enables the incipient flaws in the piping systems to be detected thus allowing for an early warning system that precludes microscopic cracking. The added safety and reliability of operation would cut down time and would result in the most efficient production of electricity.

The ability to identify certain parameters that affect the useful life of test pipes would result in an early incipient failure prediction technique and benefit the nuclear

industry. The IFD-NDE technique would enable the user to control the various failure mechanisms to his advantage. Proper use of the technique would identify the failure mechanism thus allowing different solutions to problems associated with the various failure mechanisms (stress corrosion cracking, fatigue, corrosion fatigue, and thermal shock) to be implemented.

SCOPE

The scope of this technical feasibility program includes utilization of previous investigations and engineering design data of the specific damping capacity as a nondestructive evaluation technique for testing four inch diameter, Schedule 80 stainless steel pipe. The test pipes varied, within tolerances, in the as-received condition. The laboratory data is presented in the form of design curves for the IGSCC failure mechanism. An analytical model has been developed for the field data. Moreover, a field inspection technique has been developed for field measurements.

BACKGROUND

It has been well known for over a century that materials manifest deviations from perfect elastic behavior even at small stress levels. Zener (1)* calls this behavior "annelasticity"

---

* Numbers in parentheses refer to references at the end of this report.

of materials. Since real solids are never perfectly elastic, some of their mechanical energy is always converted into heat. The various mechanisms by which this occurs are collectively termed as internal damping.

The most direct method for defining internal damping is by the specific damping capacity. Precisely, the specific damping capacity, D, is given by:

$$D = \frac{\Delta W}{W} \qquad [1]$$

where:

ΔW = the energy dissipated in one cycle, and

W = the total energy of the cycle.

The mechanism by which the energy is dissipated may be any one of the following (2):

1. relaxation by thermal diffusion,
2. relaxation by atomic diffusion,
3. relaxation by magnetic diffusion,
4. relaxation by ordered distribution,
5. relaxation of preferential distributions,
6. stress relaxation along previously formed slip bands,
7. stress relaxation across grain boundaries, and
8. stress relaxation across twin interfaces.

The phenomenon of internal damping of materials is used throughout the world to test authenticity of coins, the soundness of castings, the operating condition of railroad wheels, and the quality of musical instruments and glassware by listening to the tone and duration of sound (3). The study of internal damping has been very useful as a research tool in physical metallurgy; in vibration control of high-speed missiles, planes, and vehicles; in fatigue of materials; and in the study of properties of metals and alloys from the standpoint of theroetical and academic considerations. There are various technical terms to characterize this phenomenon: internal friction, logarithmic decrement, hysteretic constant, viscosity, elastic-phase constant, damping capacity, and so on.

The data reduction procedure and calculation of specific damping capacity that characterizes the dynamic response of the test components after a specific loading history are outlined in this section. Furthermore, the following relationships would be useful in data reduction and interpretation of results obtained on various components.

The dynamic response of the material is characterized by the damping coefficient or logarithmic decrement which can be expressed as:

$$\alpha = \left(\frac{1}{N}\right)\left(\ln \frac{A_n}{A_o}\right) \qquad [2]$$

where:

α = the damping coefficient

A_o = the vibrational amplitude of the reference cycle, and

A_n = the vibrational amplitude after N cycles.

From the determination of [2], the specific damping capacity $\left(\frac{\Delta W}{W}\right)$ for materials is calculated from the following equation:

$$\frac{\Delta W}{W} = 1-e^{-2\alpha} \qquad [3]$$

where:

ΔW = the energy dissipated in one cycle, and

W = the total energy of the cycle.

The damping coefficient (α) is determined by plotting on semi-logarithmic axes several points of the equation:

$$N_i = \ln \frac{A_{n_i}}{A_o} \qquad [4]$$

A linear relationship is established from the experimental points on the graph and the damping coefficient is determined from the linear relationship for the appropriate number of cycles. From the determination of α, the specific damping capacity $\left(\frac{\Delta W}{W}\right)$ can be determined.

Where the level recorder is used, the output signal ($A = f[t]$) is automatically translated into the function:

$$\frac{A_n}{A_o} \text{ (dB)} = f \text{ (N)} \qquad [5]$$

The determination of α is made by measuring the slope of the recorded curve. The damping coefficient is proportional to the natural logarithm of the ratio of the amplitudes. On the level recorder output, the slope of the curve is proportional to the common logarithm of the ratio of the amplitude.

EXPERIMENTAL APPARATUS, TECHNIQUES, AND PROCEDURES

The internal friction technique requires an input pulse and an associated output signal. The specific damping capacity is determined from the resultant decay curve of the output signal. In application, the technique used the input of periodic excitation pulses. The input pulses are generated by a beat frequency oscillator through a gate to a permanent magnet vibration exciter. The beat frequency oscillator supplies a continuous periodic signal at a specific frequency. The permanent magnet vibration exciter converts the electrical pulses from the oscillator into mechanical vibrations.

The vibrational response is measured by an output accelerometer. The signal from the accelerometer is filtered by a frequency analyzer and displayed on a storage oscilloscope. The specific damping capacity is determined from the decay curve of the input pulse. The accelerometer converts the vibrational response of the specific component to an electrical

signal. The output signal from the accelerometer is filtered by an audio frequency analyzer and the output decay curve is displayed on a storage oscilloscope which is externally triggered by the signal used to gate the input.

Measurements of the vibration decay are made between pulses with a pick-up transducer. On the pick-up side, the unit measures a wide range of damping levels and various degrees of amplification in time and in amplitude. The amplification units will consist of a voltage amplifier for the amplitude of the received signal and an oscilloscope for the spread in time of the received signal.

Thus, the selection of the major components has to satisfy practical requirements to allow meaningful interpretation of results. Such design criteria consists of:

1. Frequency range: 20 - 20,000 Hertz
2. High pick-up sensitivity
3. High signal/noise ratio
4. High accuracy of measurements
5. High amplification (100 dB)
6. Minimum harmonic distortion

A comparison of available off-the-shelf equipment led to the selection of several commercially available instruments. A schematic representation of the equipment including the microprocessor technology is included in Figure 1.

ACCOMPLISHMENTS TO DATE

The work accomplished under the present Nuclear Regulatory Commission program was pursued in three areas: laboratory testing at the Springlake Research Center of DAI; laboratory testing at the Battelle Pacific Northwest Laboratories; and at an active BWR facility - Commonwealth Edison Company, Dresden Unit II.

The in-house laboratory testing involved basic engineering research on the application of the IFD-NDE instrumentation to four inch stainless steel pipe. Transducer mounting orientations and techniques were investigated as were various vibrational modes and noise levels. This was accomplished by examining a 16 inch long test section of four inch 304 stainless steel pipe. The pipe section was tested under several configurations including horizontal and vertical free-standing, end supported with massive steel block, and with/without water. Flow tests were done to simulate possible noise conditions which might have to be dealt with in actual field testing. Knowledge learned of signal to noise problems and transducer mounting options during this initial laboratory phase allowed instrumentation and analysis to begin on the IGSCC pipe loop at Battelle as soon as the test loop was finished. No transition time was needed for test check out save for the normal, required, base line analysis. Figure 2 shows the DAI labortory test pipe in one of its final test configurations.

Application of the IFD-NDE instrumentation and specific damping capacity analysis began on the Battelle Northwest Laboratory Pipe Demonstration Loop in March 1979. As of 21 September 1979, four pipe samples of 304 stainless steel four inch Schedule 80 had been tested. The test section for each pipe sample was a 16 inch length in the center of the load collars (Figure 3). Located at the center of the test section was a weld. Axial fatigue loading to failure was accomplished on three of the four pipes. Table I presents the test matrix information. System parameters were as follows:

- Water temperature + 550°F (+ 288°C)
- Water flow rate 75 gpm
- Water pressure 1150 psig

Figure 4 is a photograph of the pipe test loop at the Battelle Laboratory.

Pipe Sample 1 was fatigued to 737 cycles and removed from the loop, prior to failure. IFD-NDE instrumentation and analysis had indicated specific damping capacity changes outside of a statistical confidence bandwidth as early as cycle 173. A microcrack was found in the weld/parent metal interface when the sample was cut and sectioned. Accoustic emission analysis had paralleled the IFD-NDE testing and had failed to register enough activity to qualify as crack propagation. Radiographic and ultrasonic testing also indicated no crack had

formed. Only the IFD-NDE instrumentation and analysis predicted the crack. Figure 5 shows the specific damping capacity versus fatigue cycles which indicated early crack formation. Data from other frequencies and accelerometer locations are similar in appearance.

Pipe Sample 2 was fatigued to 246 cycles and failure. Crack initiation and growth was indicated by the IFD-NDE technique around cycle 37. No accoustic emission data was reported on this test. Figure 6 is a representative plot of specific damping capacity versus fatigue cycles for pipe Sample 2.

Pipe Sample 3 was fatigued to 76 cycles and failure. The applied load was 150 percent of field strength. Crack initiation and growth was indicated by the IFD-NDE technique at cycle 12. Figure 7 is a photograph of the failed pipe section.

Pipe Sample 4, was fatigued to 163 cycles and fracture. Fatigue loading was again 136 percent of yield. The IFD-NDE technique indicated crack initiation and growth at cycle 142.

Pipe sample five was fatigued utilizing a cyclic load after a number of step loads were ap lied to the pipe. The initial step load was applied to the pipe sample at 100 percent of the 550°F yield stress. Incremental step loads increasing by 10 percent per 24 hour period were applied to

the pipe sample. IFD data was obtained on the pipe samples each one hour period of the 24 hour period where the stress being applied to the pipe was held constant. Upon initiation of the 140 percent load based upon a 550 yield stress the pipe was then loaded in a cyclic manner. The pipe sample was then fatigued to a through wall crack culminating in a leak in the pipe.

Additionally, nine pipe samples were welded at the Battelle Pacific Northwest Laboratory for use in the DAI pipe fatigue loop. An additional seven pipe samples were welded that would be fatigued in the PDL at the Battelle facilities.

Concurrent to the mechanical loading tests to be performed on the 16 pipe samples, DAI has initiated work that will provide a computer software package that will automate the IFD-NDE technique. Presently the computer software package has been able to obtain and analyze data from the in-house tests performed at DAI's Springlake Research Center.

Parallel to the Battelle pipe tests, instrumentation of feedwater sparger safe ends was accomplished. This occurred at the Commonwealth Edision Company Nuclear Generating Facility, Dresden Unit II, located outside Morris, Illinois. Feedwater lines C and D were instrumented, employing strap mounts similar to those used for the Battelle laboratory tests and developed during the in-house tests at the DAI laboratory.

Figure 8 shows a photograph of the input transducers mounted on the feedwater lines. The transducers are partially obscured by the large reflecting insulation used on the reactor piping.

In addition to the above identified instrumentation placed at Dresden's Nuclear Facility, two 4 inch bypass lines have been instrumented and the ability to generate data exists at that facility. The instrumentation consists of the input and output transducers being attached directly to the four inch bypass lines and an instrumentation package that can be manually placed in the control room of the facility thus enabling the data to be collected.

The feedwater lines of the once through steam generators at the Zion Nuclear Facility has also been instrumented. The addition of this work to the program would enable a larger data base to be gathered.

REFERENCES

1. Zener, C., "Internal Friction in Solids, Theory of Internal Friction in Reeds," Physical Review, Volume 52, page 230, 1937.

2. Zener, C., "Elasticity and Anelasticity of Metals," The University of Chicago Press, Chicago, 1948.

3. Jenson, J. W., "Damping Capacity - Its Measurement and Significance," Bureau of Mines Report of Investigation 5441. United States Department of the Interior, 1959.

TABLE I

IGSCC PIPE SAMPLE TEST MATRIX

PIPE SAMPLE	FATIGUE LOAD (% YIELD STRENGTH)	SENSITIZED	PIPE SEPARATION (CYCLES)
1	135	YES	NO
2	135	YES	YES (246)
3	150	YES	YES (76)
4	136	NO	YES (163)

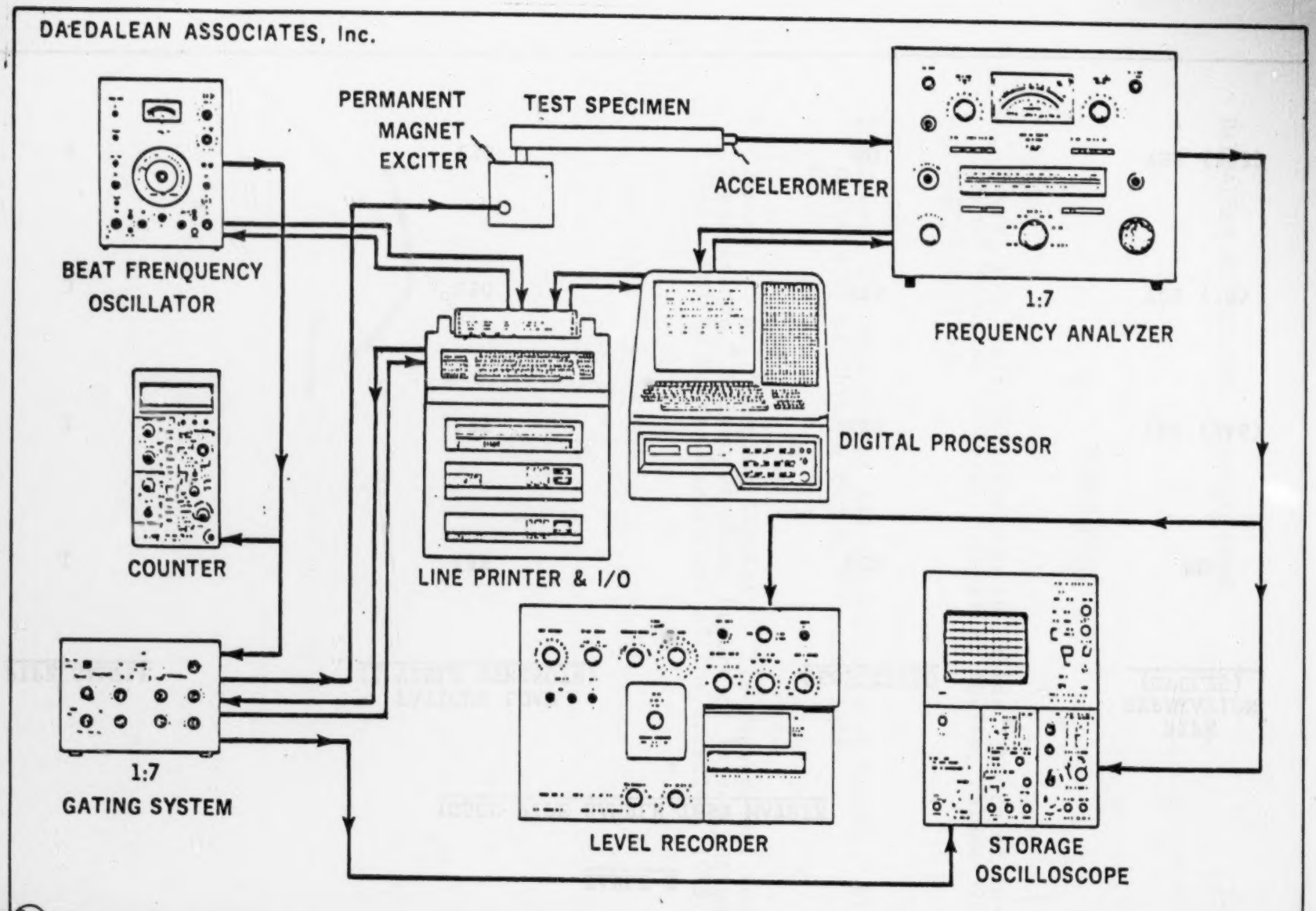


FIGURE 1 NDE SYSTEM CONFIGURATION UTILIZING A COMPUTER ASSISTED DIGITAL PROCESSOR FOR PREDICTING INCIPIENT FAILURE

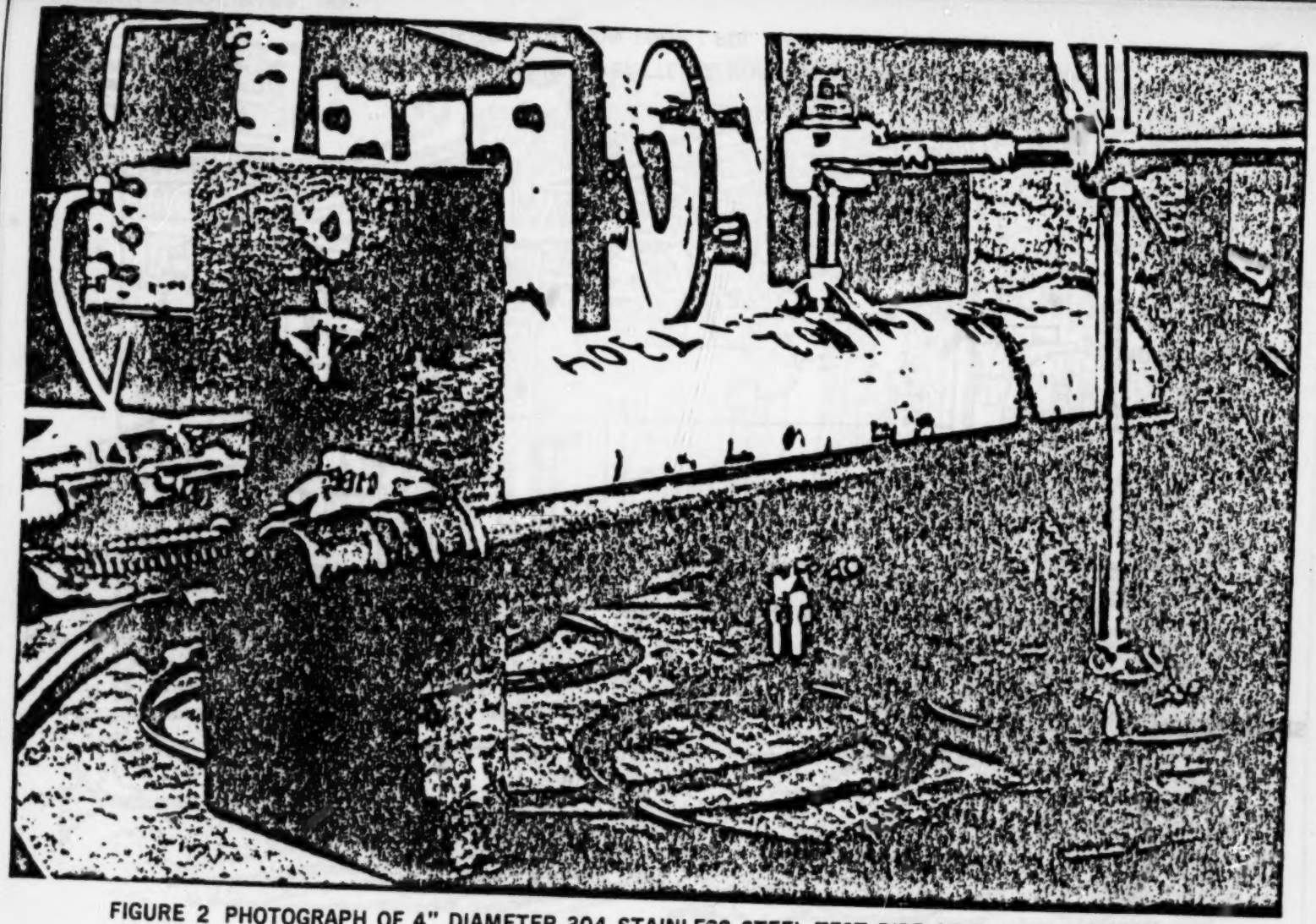


FIGURE 2 PHOTOGRAPH OF 4" DIAMETER 304 STAINLESS STEEL TEST PIPE AT THE SPRINGLAKE RESEARCH CENTER LABORATORY OF D.A.I.

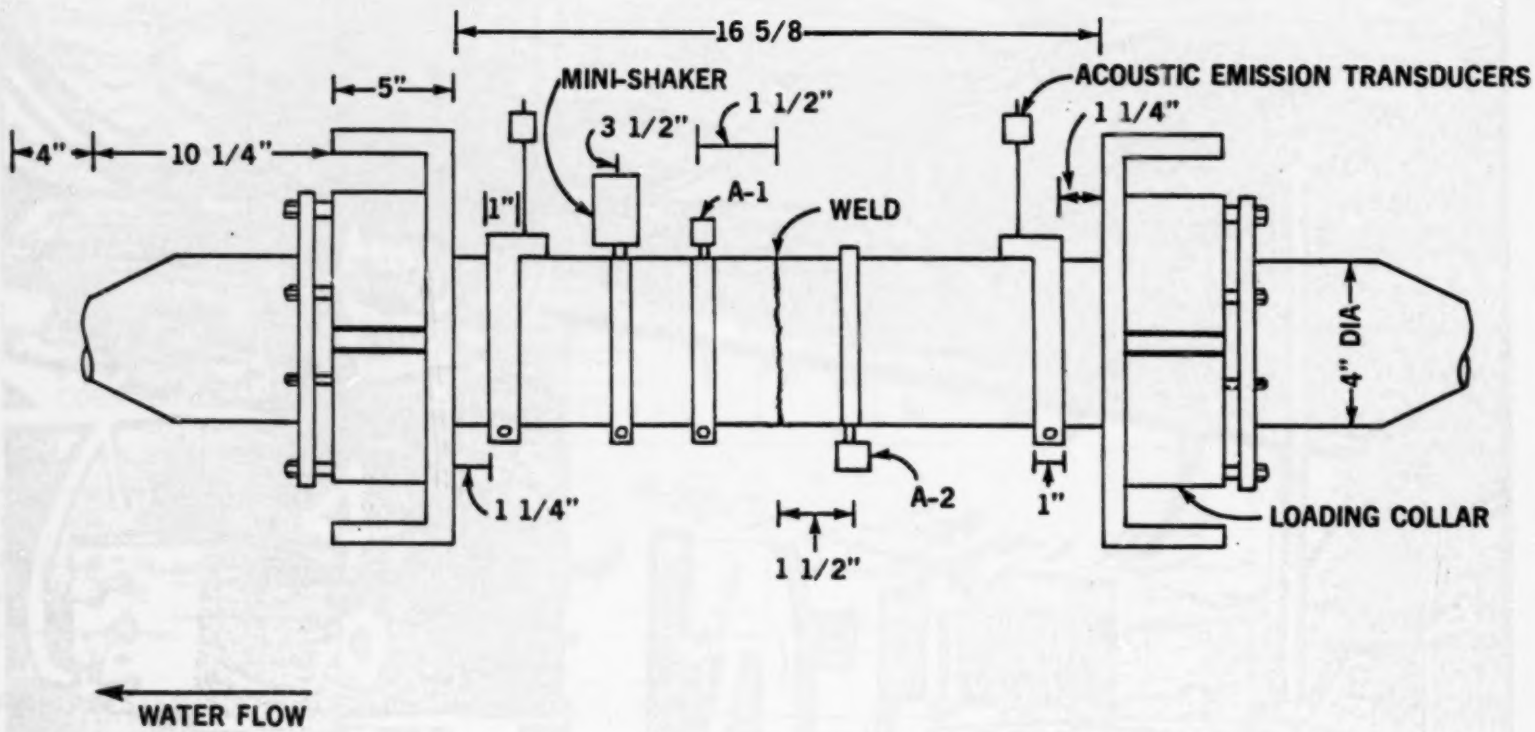


FIGURE 3 SCHEMATIC OF BATTELLE NORTHWEST LABORATORY PIPE DEMONSTRATION LOOP (PDL) PIPE TEST SECTION

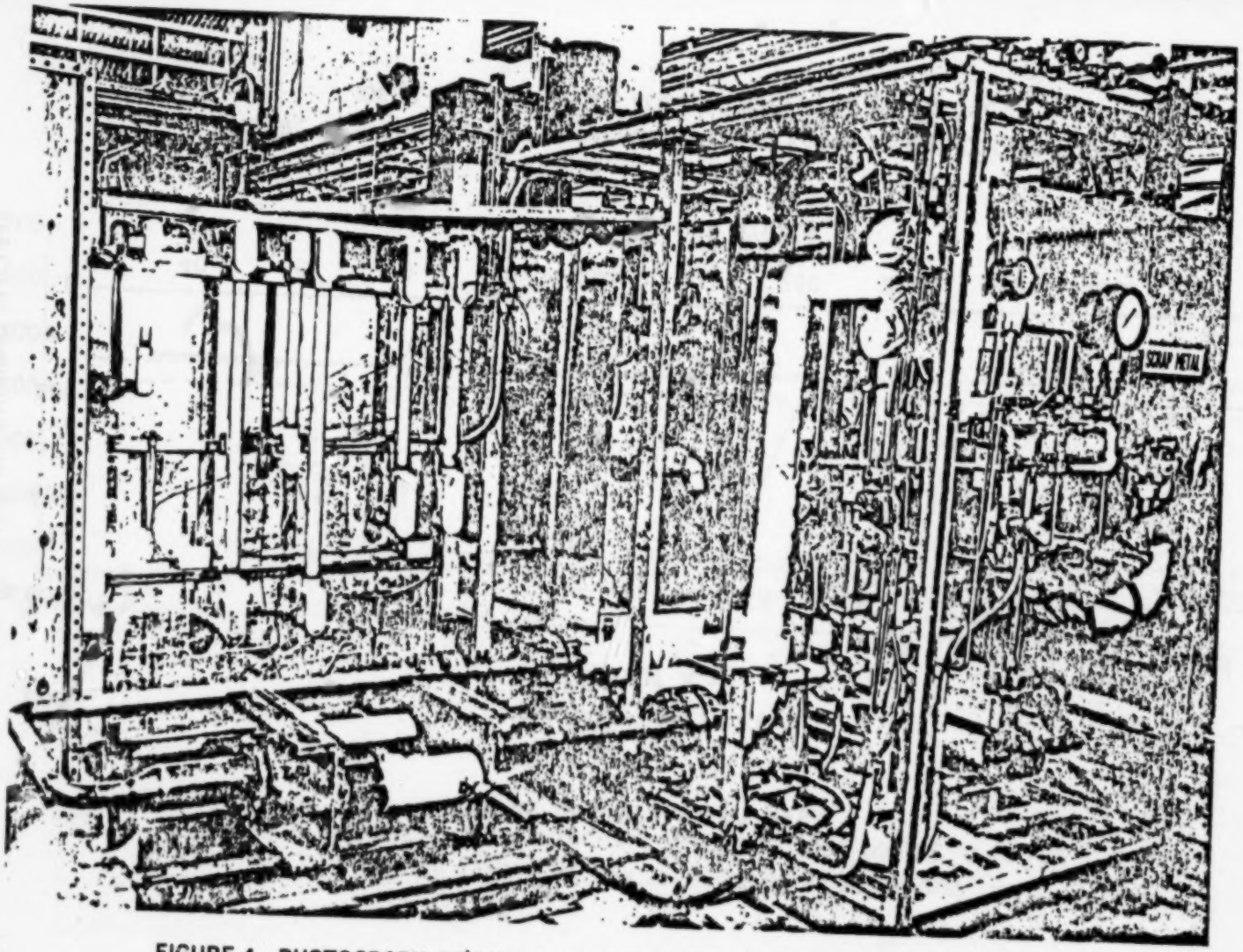


FIGURE 4 PHOTOGRAPH OF (PDL) AT THE BATTELLE NORTHWEST LABORATORY

DAEDALEAN ASSOCIATES, Inc.

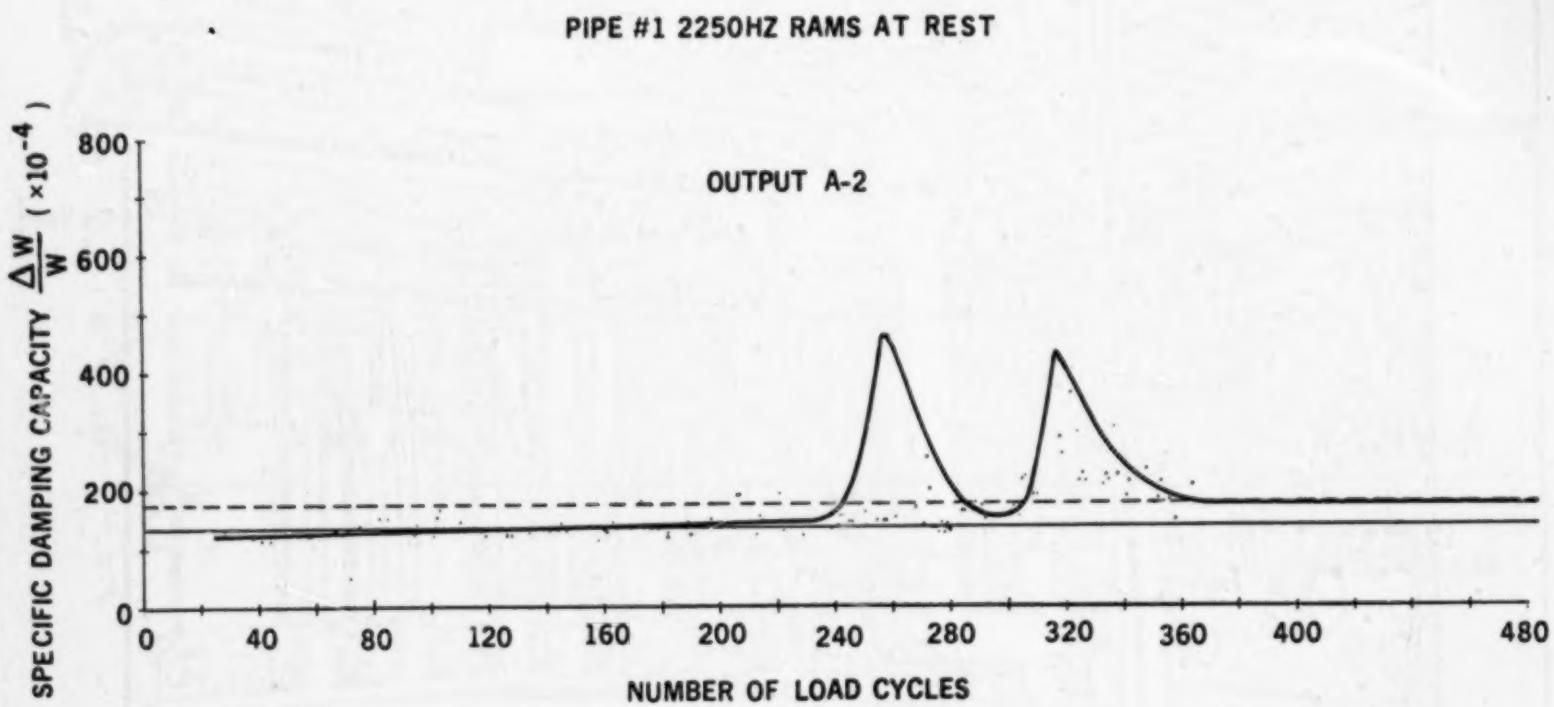


FIGURE 5 SPECIFIC DAMPING CAPACITY VERSUS FATIGUE LOAD CYCLES FOR IGSCC PIPE SAMPLE #1. ACCELEROMETER A2 AT 2250HZ

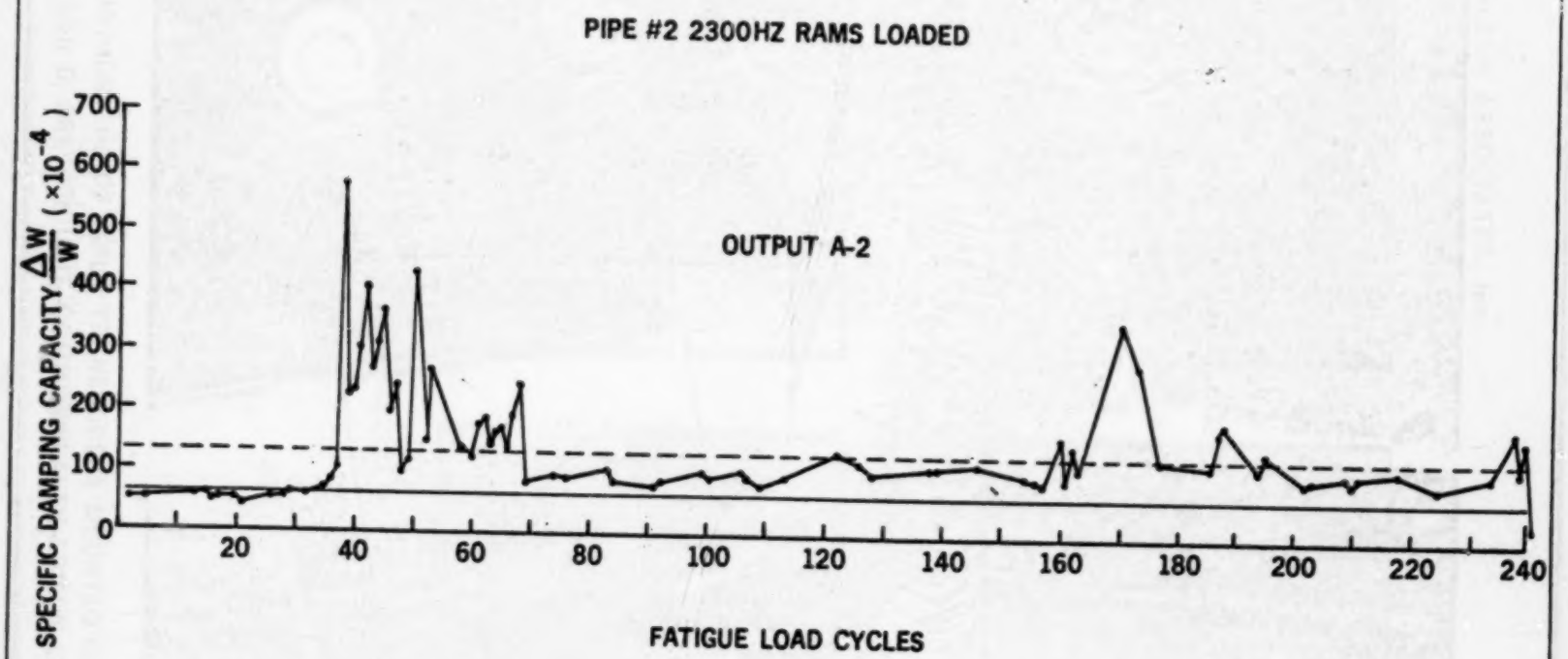


FIGURE 6 SPECIFIC DAMPING CAPACITY VERSUS FATIGUE LOAD CYCLES FOR IGSCC PIPE SAMPLE #2. ACCELEROMETER A2 AT 2250HZ

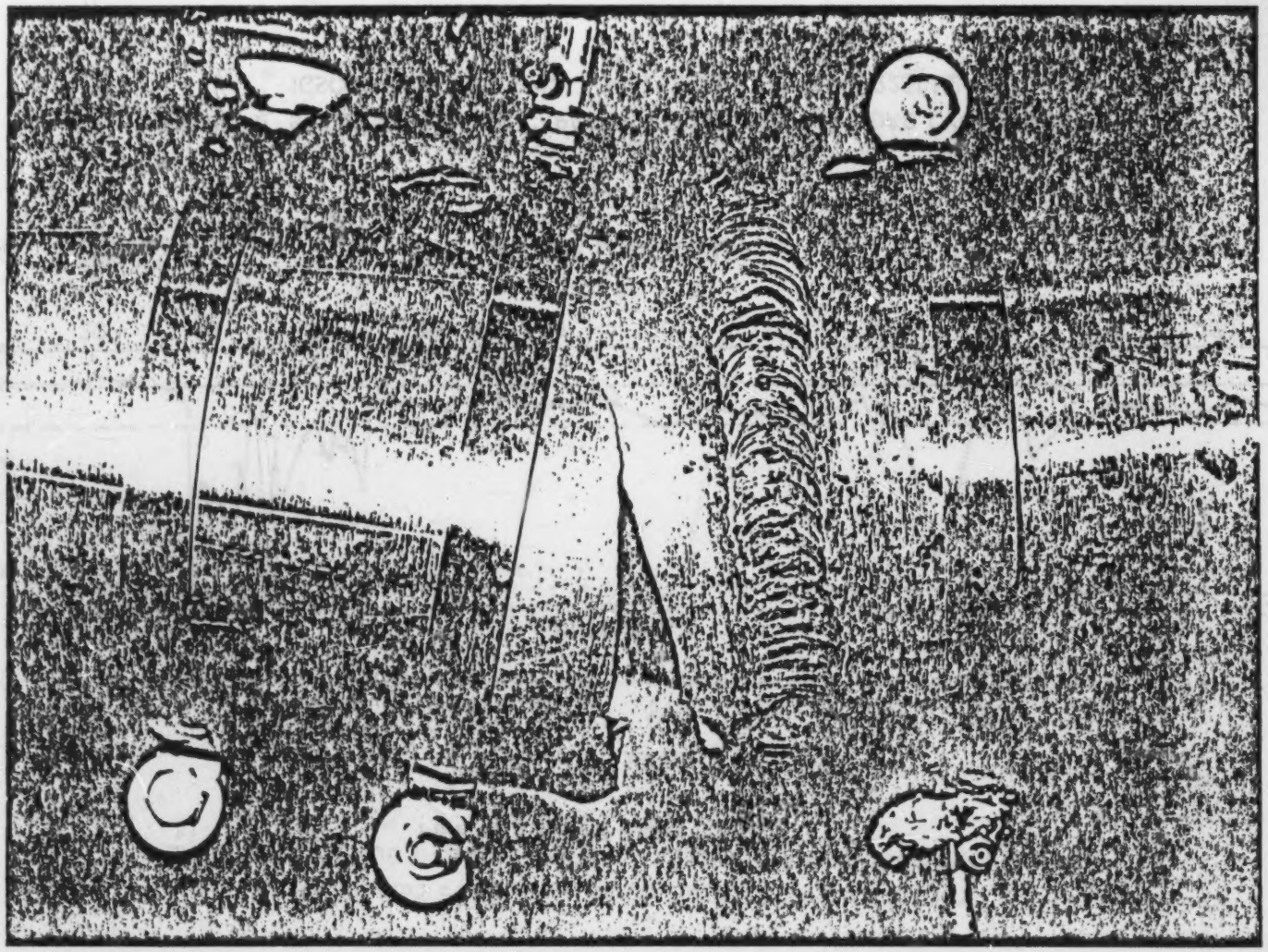
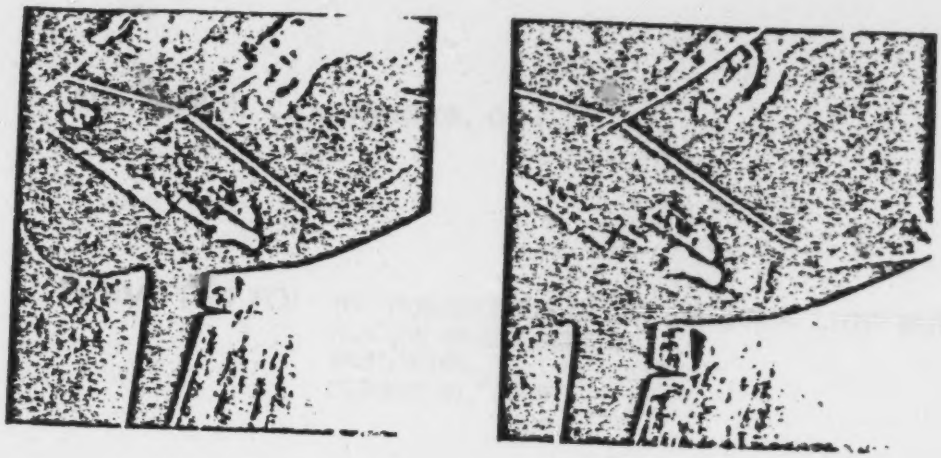
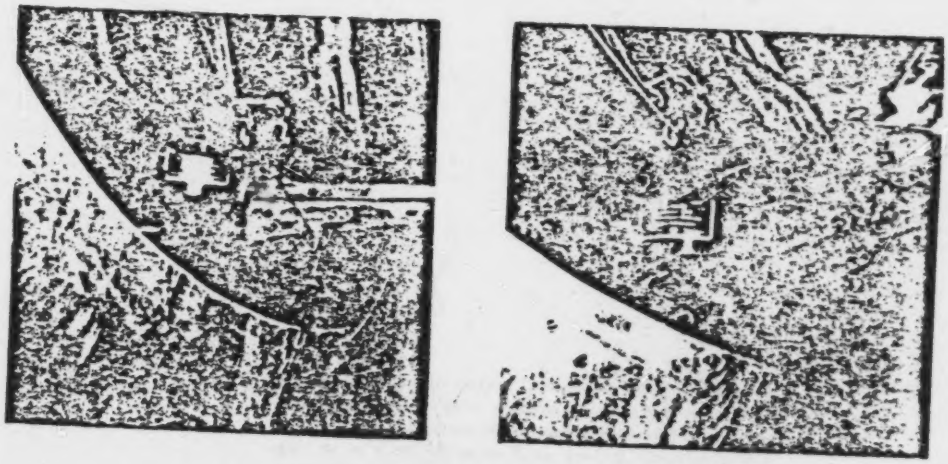


FIGURE 7 PHOTOGRAPH OF FAILED PIPE SECTION. IGSCC PIPE SAMPLE #3



C FEED NOZZLES



D FEED NOZZLES

FIGURE 8 PHOTOGRAPHS OF INPUT TRANSDUCERS MOUNTED ON FEEDWATER LINES C AND D, DRESDEN UNIT, COMMONWEALTH EDISON COMPANY

LA-UR-80-2858

TITLE: CATEGORY I STRUCTURES - SAFETY MARGIN AT ULTIMATE LOAD

AUTHOR(S): C. A. ANDERSON, Q-13

SUBMITTED TO: FOR PRESENTATION AT THE 8TH ANNUAL LIGHT WATER REACTOR SAFETY MEETING
WASHINGTON, DC
OCTOBER 31, 1980

By acceptance of this article, the publisher recognizes that the U.S. Government retains a nonexclusive, royalty-free license to publish or reproduce the published form of this contribution, or to allow others to do so, for U.S. Government purposes.

The Los Alamos Scientific Laboratory requests that the publisher identify this article as work performed under the auspices of the U.S. Department of Energy.

University of California

LOS ALAMOS SCIENTIFIC LABORATORY
Post Office Box 1663 Los Alamos, New Mexico 87545
An Affirmative Action/Equal Opportunity Employer

Form No. 836 R3
St. No. 2629
12/78

UNITED STATES
DEPARTMENT OF ENERGY
CONTRACT W-7405-ENG. 36

CATEGORY I STRUCTURES - SAFETY MARGIN AT ULTIMATE LOAD

C. A. Anderson
Los Alamos Scientific Laboratory

Recently, strength and failure mode assessments of existing nuclear plant structures have become necessary because of new or additional design criteria or safety considerations. These assessments, to be realistic, must account for the structural behavior in the nonlinear regime and up to complete collapse of the first element. Because of the complexity of the analyses and the lack of documented structural system behavior, structural analysts have in the past generally avoided considering the nonlinear regime.

By using modern finite element codes, it is now possible to theoretically predict structural behavior in the nonlinear regime; however, the experimental verification of nonlinear structural behavior is lacking. The lack of experimental verification was underlined in a report prepared by the Committee on Earthquake Engineering Research (a division of the National Academy of Engineering) in 1969. This thought was emphasized by the statement: "The principal difficulty in the analytical process stems from the lack of information concerning the problem characteristics, namely the form of the earthquake ground motion and the physical properties of the structural system....The basic definition of the dynamic properties of the structure...can only be accomplished by experimental methods....In practice, however, it will be necessary to carry out an integrated test program in which materials, elements, assemblages, and complete systems are tested."[1]

In 1975, an Electric Power Research Institute report prepared by Applied Nucleonics Company, Inc., made the same point after a study of the aseismic design of nuclear power plants. This report emphasized the need for verifying the current aseismic design methods: "It is vital that parallel experimental programs be supported to verify the accuracy of analytical tools. Such verified analytical capabilities could be extremely useful in determining remaining capacity of a nuclear power plant subjected to damaging seismic excitation...."[2]

The objective of the program described in this paper is to supply test information needed to assess the structural capability of Category I structural systems that are loaded beyond their design limits. The program activities were started in the latter half of FY 1980. The activities for FY 1981 are to continue the development of structural test plans, initiate design, construction and testing of small structures, continue to develop and perform analytical prediction of limit states, and to complete a US Reactor Concrete Structural Technology Report. In FY 1982, the testing of concrete structural systems will take place along with analytical modeling of selected experimental tests.

Technical achievements of the Category I Structures Program by the end of FY 1980 include the following:

(i) Visits to and conversations with representatives of architectural engineering firms have been carried out to obtain information on Category I

structures and to identify procedures used by such firms in the design of Category I structures. Bechtel Corporation, TVA and Sargent and Lundy were contacted in regard to this task. Particular concerns were voiced over damping values for cracked shear walls and their influence on ultimate load behavior and modeling of interaction effects of structural elements with equipment and piping. Detailed plant drawings were obtained in some cases.

(ii) A peer review panel for the Category I Structures Program has been organized and met in September in Knoxville, TN. Although concern was expressed about the use of scale models to establish the margin to failure of Category I structures, the review panel did agree that damping values, effect of duration of motion, and stiffness degradation could be investigated in test structures of relatively small size.

(iii) A review of seismic test facilities was carried out with regard to availability, specimen size, test frequency range, and force limitations. A trip was made to the Earthquake Engineering Research Center to assess the capabilities for seismic testing of Category I structures.

(iv) The use of scale models for determining behavior of prototype structural elements near failure has been considered. Scaling laws for the ultimate strength structural model have been derived, which include proper accounting of gravitational forces. The difficulty of employing true scale models in the test program has been identified.

(v) Based on the suggestions of the peer review panel and the representatives of the AE firm, and the review of the capabilities of seismic test facilities, a preliminary test plan has been formulated that uses small test structures (not true scale models) representing a multi-story Category I building. Elastic modes of vibration have been calculated using finite element models as well as predictions of the ultimate load. Aspect ratios for the shear walls have been determined to ensure that a shear failure occurs rather than a flexure failure.

A parallel program at the Los Alamos Scientific Laboratory is the Containment Buckling Program. In this program the behavior of thin-walled steel containment shells near ultimate load is examined analytically and experimentally, but here the failure mode is buckling and ensuing collapse. Progress during FY 1980 in this program has included buckling tests on thin-walled cylindrical shells with reinforced circular openings to verify the ASME area replacement rule.

The objective of both of these programs is to provide completely documented experimental test data on ultimate load behavior of small test structures that can be used to qualify (or develop) computer codes. These codes can then be used with more confidence to predict ultimate load behavior of prototype structures. Many Category I structures and structural systems provide the last line of defense for protection of the public in the case of beyond-design-basis accidents. Hence, knowing the margin to failure of these structures will give safety factors on existing Category I structures that can be confidently used in licensing nuclear plants for abnormal loads or for relicensing for revised design criteria.

REFERENCES:

1. G. W. Housner, et. al., "Earthquake Engineering Research," National Academy of Sciences Report, p. 186 (1969).

2. G. E. Howard, P. Ibanex, and C. B. Smith, "Seismic Design of the Nuclear Power Plants - An Assessment," Applied Nucleonics Company, Inc. report, prepared for EPRI, 273, June, 1975.

Structural Margins to Failure

- CATEGORY I STRUCTURES -

Program Objectives

To supply experimental information needed to assess the structural capacity of Category I structures that are loaded beyond their design limits.

SMtoF – CATEGORY I STRUCTURES –

Program Specifics – phase 1

- Classification of Category I structures
  - survey of A/E firms
- Survey of F.E. codes for the post-elastic analysis of R/C structures
- Formulation of initial test plan for representative structures

SMtoF – CATEGORY I STRUCTURES –

Program Specifics – phase 2

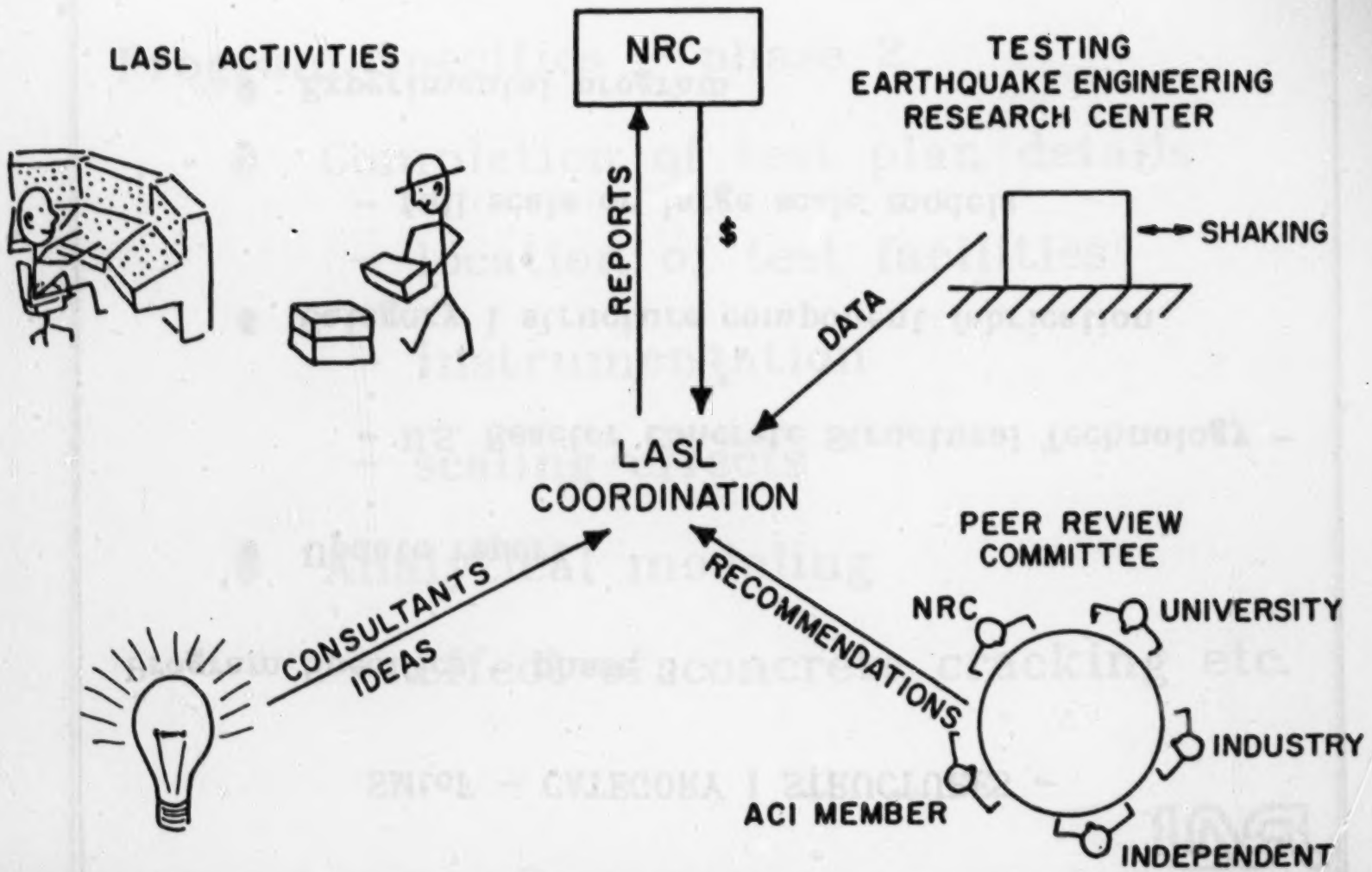
- Completion of test plan details
  - location of test facilities
  - instrumentation
  - scaling effects
- Analytical modeling
  - effect of concrete cracking etc.

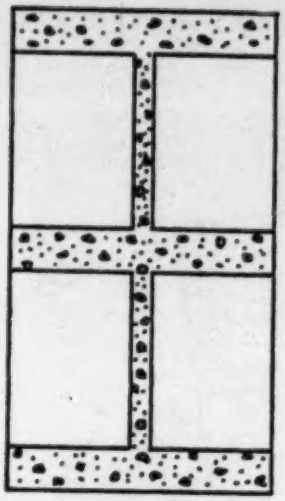
SMtoF – CATEGORY I STRUCTURES –

Program Specifics – phase 3

- **Update report**
  - **U.S. Reactor Concrete Structural Technology –**
- **Category I structure component fabrication**
  - **full scale or large scale models**
- **Experimental program**
  - **testing to failure**

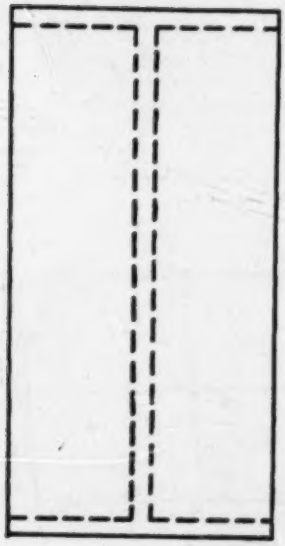
STRUCTURAL MARGINS PROGRAM





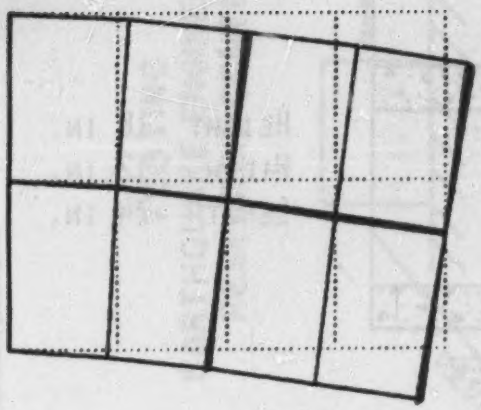
HEIGHT ~16 IN.
WIDTH ~12 IN.
LENGTH ~24 IN.

END VIEW

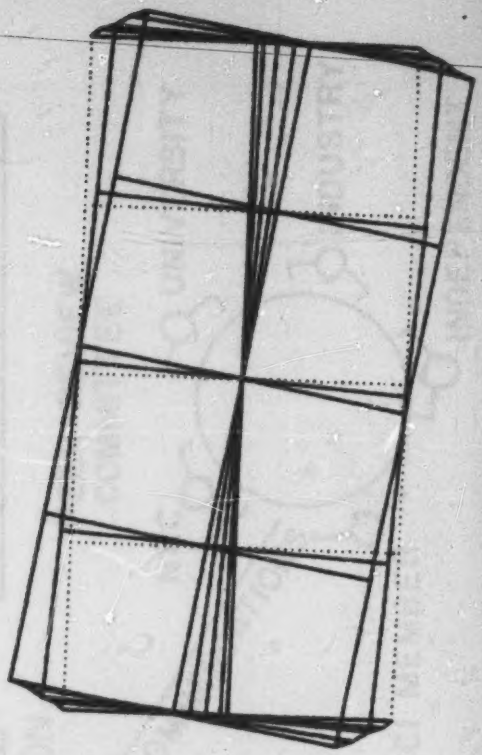


TOP VIEW

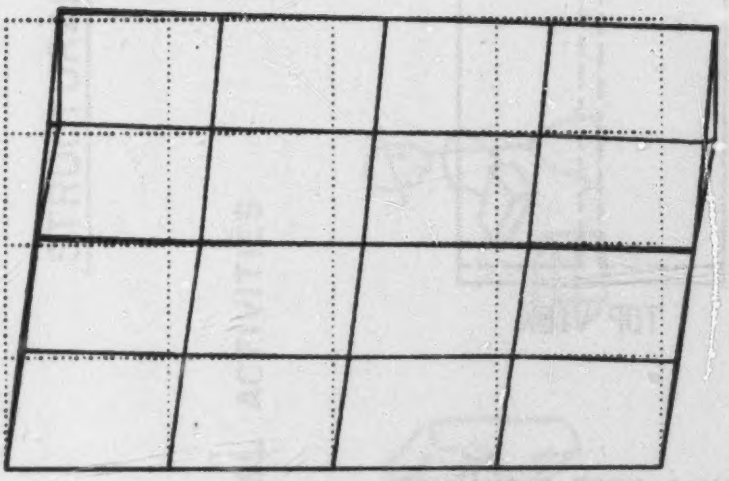
SHEAR WALL TEST STRUCTURE



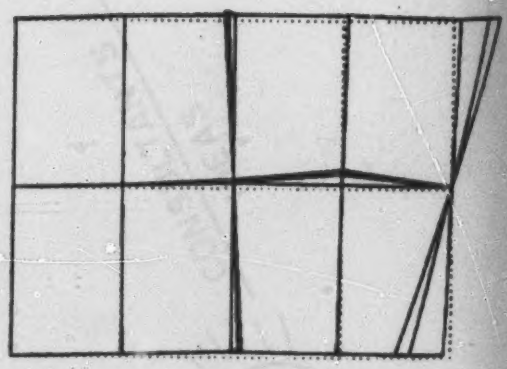
Sidesway mode.



Torsional mode.

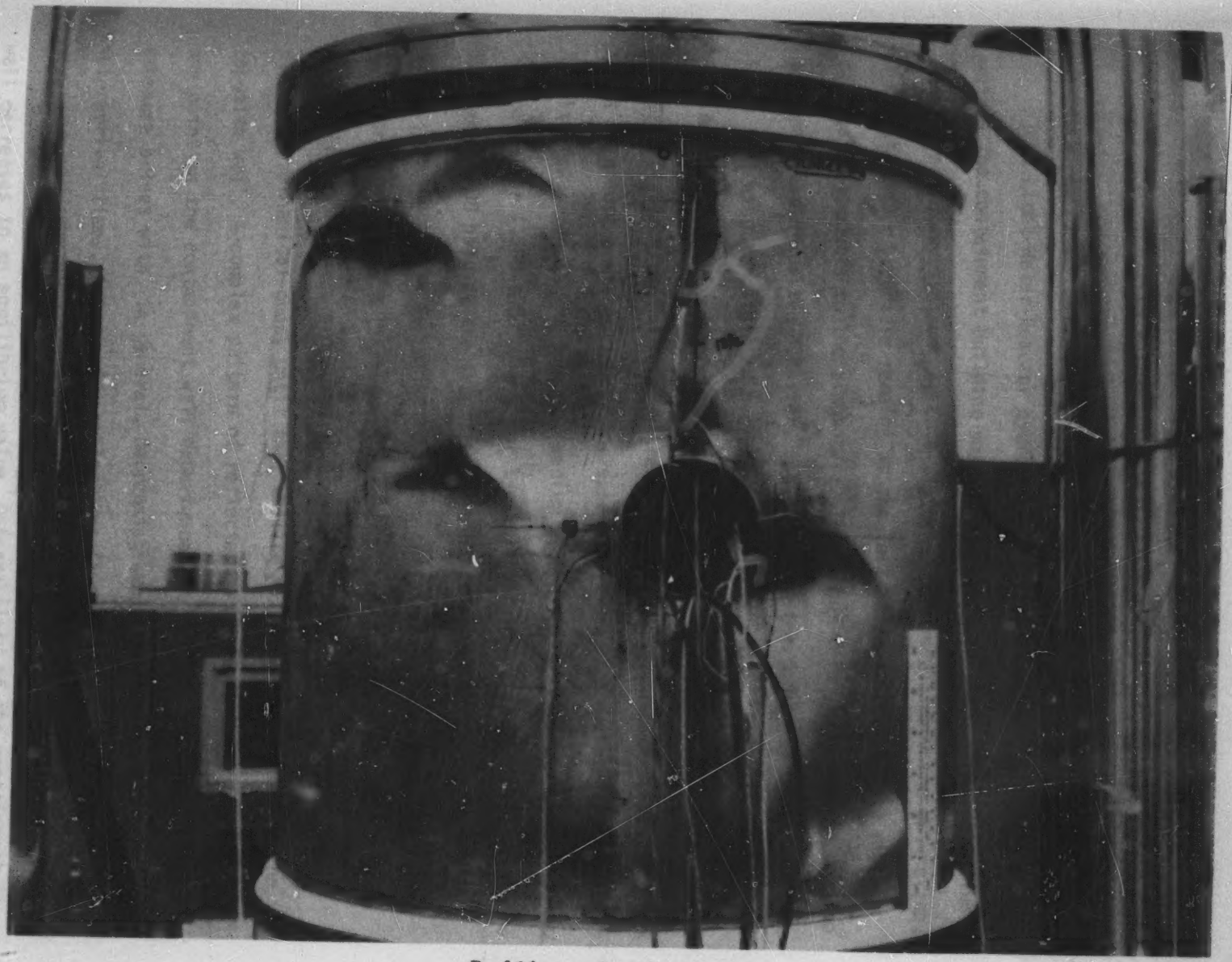


Shearing mode.



Roof rocking mode.

Vibrational modes of shear wall test structures.



Buckling mode of cylinder
with circular cutout.

Hydrogen Explosion

by Michael N. Fardis, M.I.T.

(Project: "Containment Safety Under Internal Explosions" 9-1-80 - 2-28-82)

In a Light-Water-Reactor hydrogen accumulation above a certain limit within the containment building may lead to hydrogen detonation. As failure of several safety systems and core dispersal may result from the detonation, the containment structure remains as the last defense line against release of fission products to the atmosphere. The aim of this project is to develop an understanding of the behavior, and to assess the integrity of the containment, following a postulated hydrogen detonation.

A detonation causes a shock wave which is driven and supported by the energy released from the chemical reaction of hydrogen with oxygen. The shock front is almost a discontinuous increase in pressure, temperature and density, and it propagates in the unburned medium with a speed exceeding that of sound in the latter (1500-2000 m/sec. for typical containment conditions). Pressure may exceed 200 psig at the front, and behind it it decays fast to a steady value. The shock wave is reflected on the containment wall, inducing a very large pressure (2 to 3 times the incident pressure, for normal incidence) on it. Reflected waves propagate inwards in the burned gas (implosion) as purely mechanical ones, until they collide with those reflected on the opposite wall. A complex pattern of reflected shock waves is created, and any point on the containment wall experiences an irregular series of pressure pulses with, in general, decreasing amplitude and increasing interarrival times. After a few reflections (i.e., after a few tenths of a second), the burned gas reaches its steady state and the pressure on the wall decreases to an equilibrium value. Deformability of the containment is, in general, beneficial, as it decreases peak pressures on the wall and leads

to some energy dissipation by plastic deformations.

The response of the containment to the shock depends on the distribution in space and the dependence on time of the pressure on the wall. This latter pressure is rich in high frequency components, which excite higher vibration modes if the corresponding mode shapes match closely the spatial distribution of wall pressure for a large portion of the time-history. Because of the relatively short duration of the series of pressure pulses, extreme response will occur after the wall pressure decays to its steady value. For those detonations of interest considerable inelastic action will take place, making necessary a nonlinear dynamic analysis in the time-domain. Although geometric nonlinearities (due to finite strains and/or large displacements and rotations) may become significant near "failure", material nonlinearities are important from the beginning of the response. Limiting attention to R/C or P/C containments, steel inelasticity and concrete cracking introduce significant nonlinearities. The latter occurs in all three directions (vertical-radial, vertical-in plane and horizontal) and it affects very much the interaction between the steel components (liner, rebars and tendons, if any) in bending, and the shear deformations of the wall. (Shear stiffness is provided mainly by aggregate interlock and dowel action). Response prediction is very sensitive to how well these effects are modeled.

Failure is by exhaustion of rebar or tendon ductility, (i.e., rupture), either at the upper part of the cylinder or at the dome. These regions are subjected to extensive bending action for which they have not been designed, and they have no seismic reinforcement as a reserve. The liner does not tear before the main reinforcement breaks, because of its inherent high ductility (which is further increased by the high accident temperature). If

mechanical splices in the rebars have lower(dynamic) ultimate strength than the rebar material, they will be the first to fail, and failure strain will be the rebar strain at a stress equal to the splice ultimate strength. Otherwise, failure strains will be equal to the elongation at failure of the rebar material, resulting in a large increase in overall ductility. Rupture of one or a few rebars or tendons within the most critically stressed region will precipitate similar failures in the vicinity, therefore containment resistance is sensitive to the lower tail of the statistical distribution of ultimate strengths for splices, rebars or tendons.

DURING ACCIDENT THAT INVOLVES CORE HEAT-UP

$$\left.\begin{array}{l} \text{Zr (in fuel cladding)} + H_2O \\ \text{Fe (reactor vessel \& internals)} + H_2O \\ H_2O + \text{radiation} \\ \text{Zn (in wall paint)} + H_2O \end{array}\right\} \rightarrow H_2$$

PROBLEM:

Depending on composition, a mixture of H_2, air & steam may <u>burn</u> or <u>detonate</u>.

If H_2 > lower flammability limit: Deflagration.

Quasistatic increase in p, T. Slow decay thereafter.

Possible containment overpressarization

if $p_{burning} + p_{steam}$ > ultimate capacity.

If H_2 > lower detonation limit: Detonation.

Shock wave with supersonic speed.

Possibly failure of several Eng. Safety Systems

and core dispersal. Then:

Containment: Last defense line. Will it survive the shock?

In between: Quasi-detonations. Shock waves in front of flame front.

Large uncertainties in detonation phenomenon and assoc. limits.

Extrapolation of small scale, clean experiments to

containment geometry: Questionable.

It seems: In small-volume containments (Mark III, Ice condenser),

detonation possible.

In large-volume containments (e.g., PWR dry), detonation

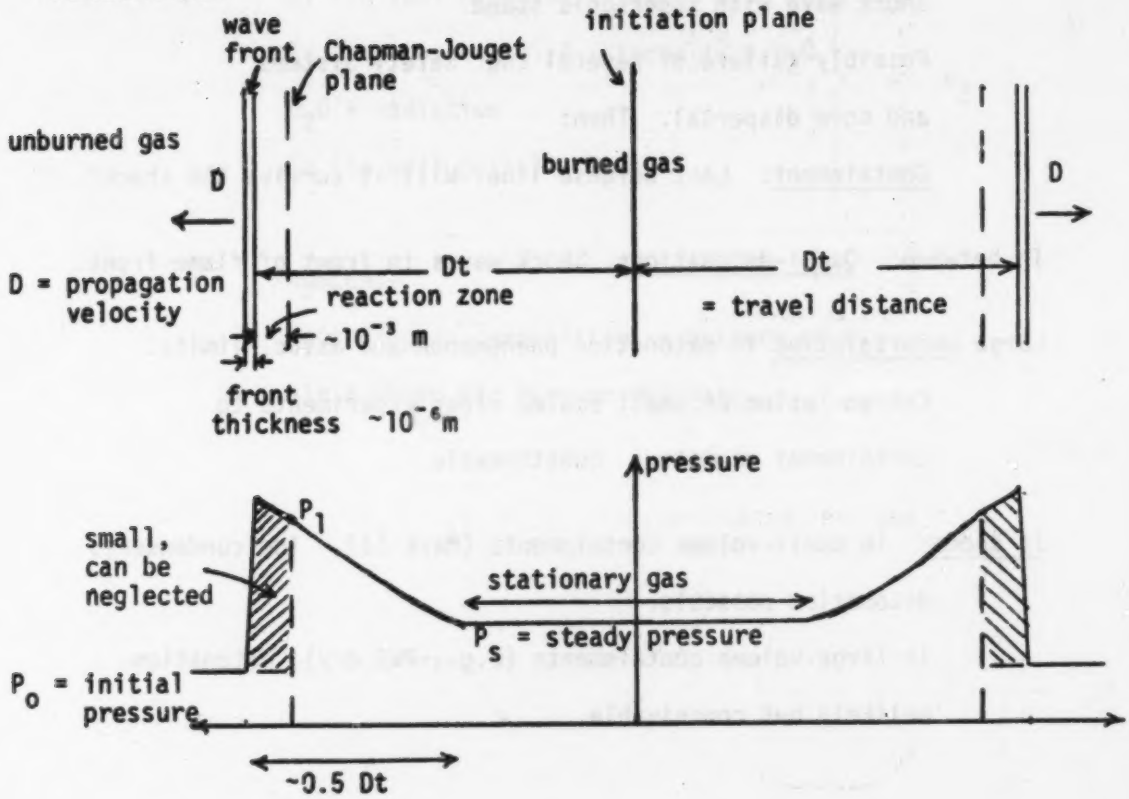
unlikely but conceivable.

DETONATION: Shock wave is driven by energy from chemical reaction.

Shock front: abrupt increase in p, T, density.

Propagation velocity > 2000-m/sec. Peak pressure ~ 100-300 psig.

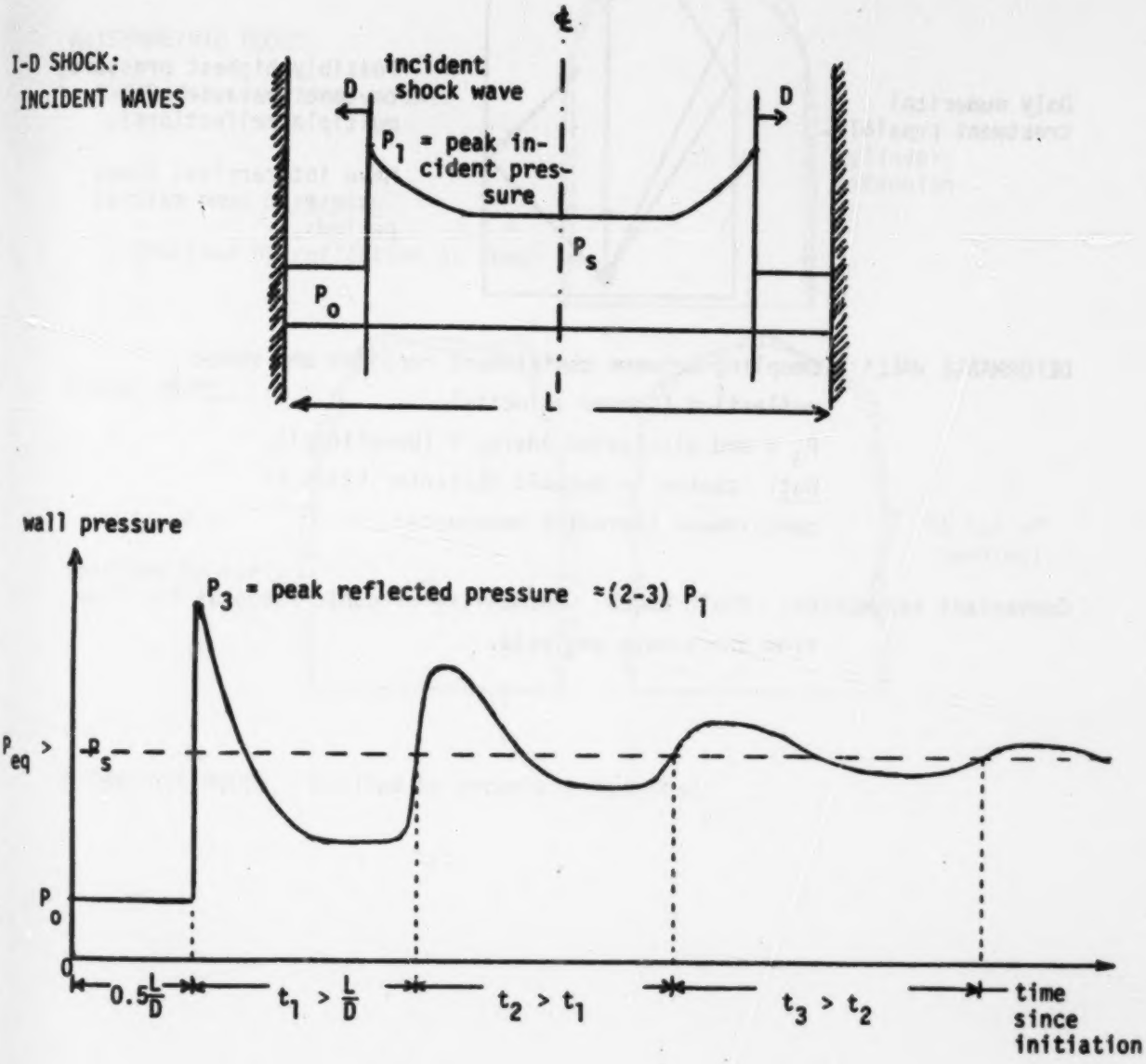
PLANAR SHOCK



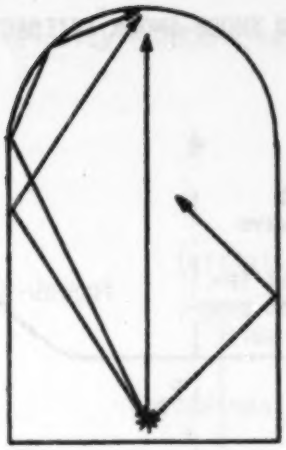
CYLINDRICAL OR SPHERICAL SHOCK.

Behind front pressure decays much faster than in planar case.

SHOCK-WALL AND SHOCK-SHOCK INTERACTION.



ACTUAL GEOMETRY:



Only numerical treatment possible.

Possibly highest pressures on dome (because of multiple relfections).

Wave interarrival times ~ close to some natural periods.

DEFORMABLE WALLS: Coupling between containment response and shock reflection (Common velocity).

P_3 ↓ and dissipated energy ↑ (Beneficial).

<u>But</u>: change in dynamic characteristics of containment (possible resonance).

Convenient assumption: RIGID WALLS. Uncoupling of cont. response from shock-wave analysis.

LINEAR ELASTIC CONTAINMENT DYNAMIC RESPONSE

AXISYMMETRIC MODES:

Excited by ignition on axis

<u>1st mode</u>

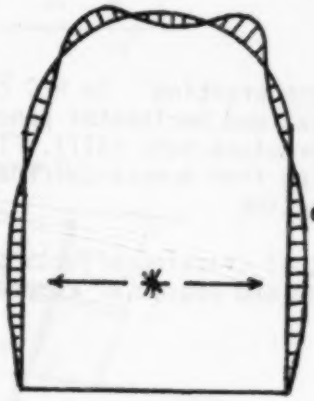
(excited by reflection at dome)

cylinder extension

<u>Higher modes.</u>

e.g.

(excited by early wall reflection)



or

(a lot of bending).

ASYMMETRIC MODES: Excited by eccentric ignition.

NONLINEAR DYNAMIC RESPONSE

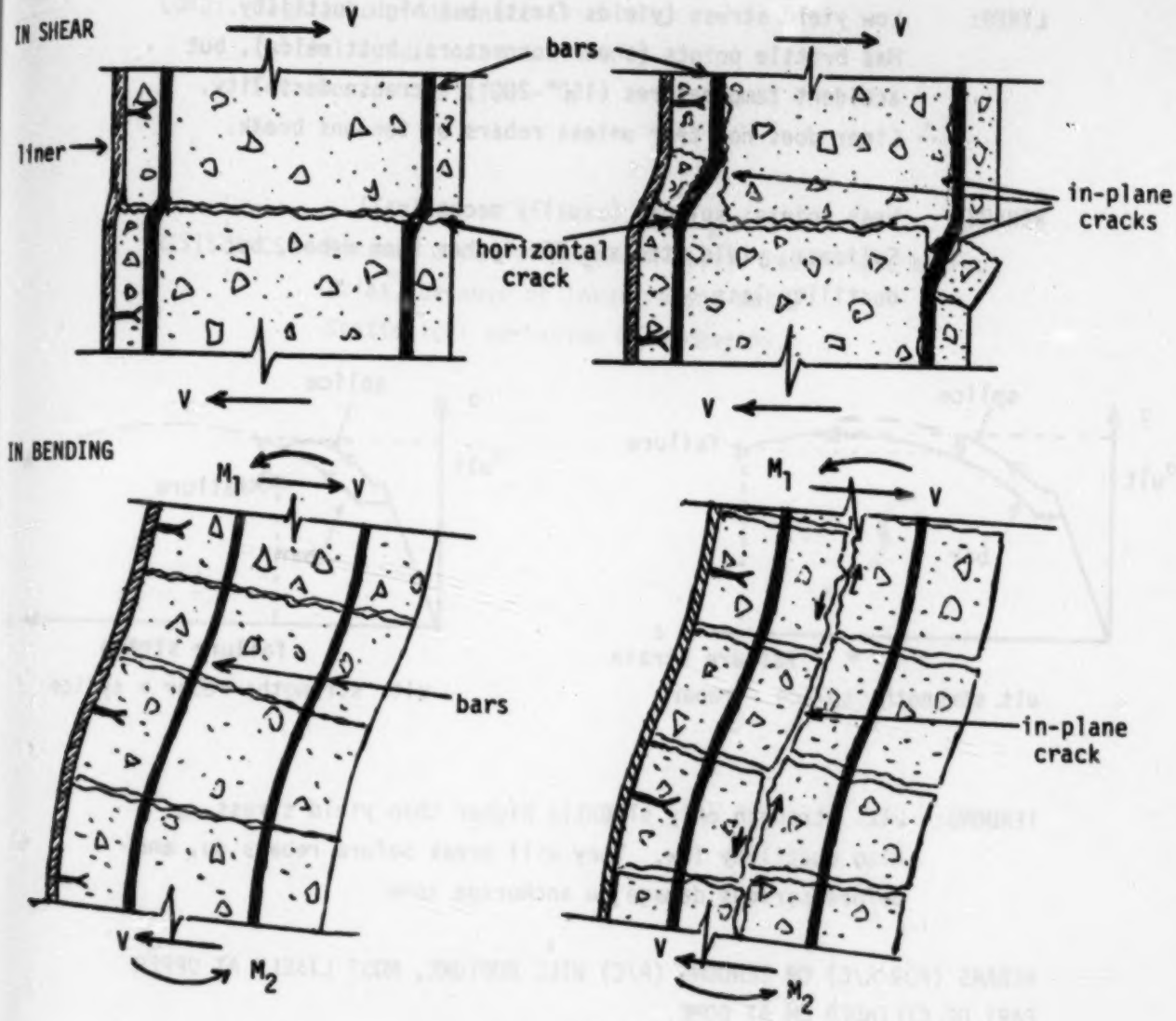
(Integration of Eqs. of Motion in the t-domain)

NONLINEARITIES

- GEOMETRIC: Finite strains, large displacements.
- MATERIAL
  - Steel yielding and strain-hardening. Need high-strain-rate constitutive relations.
  - Concrete cracking: In R/C containments, vertical and horizontal penetrating cracks from pressure test (SIT). In-plane cracking from across-thickness wave propagation.

    Extent of cracking affects shear deformations and degree of component interaction.

EFFECTS OF IN-PLANE CRACKING ON WALL DEFORMATIONS

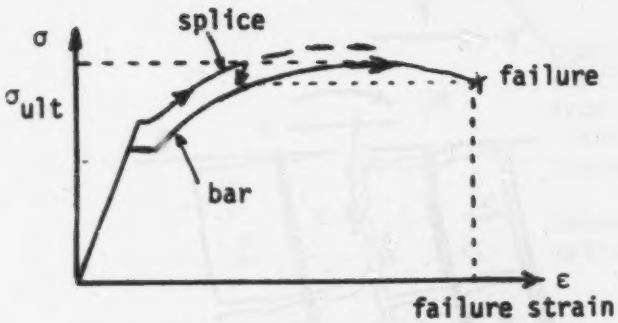


FAILURE MODES

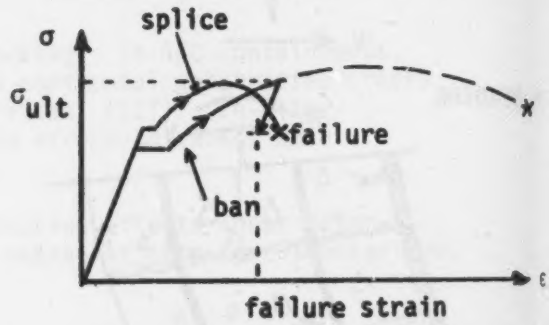
LINER: Low yield stress (yields first) but high ductility. Has brittle points (shear connectors, butt welds), but accident temperatures (150°-200°) increase ductility.

∴ Liner does not tear unless rebars or tendons break.

REBARS: Weak points: splices (usually mechanical).
Splices: yield stress ≥ 25% higher than rebar, but ductility less.



ult. strength: splice > rebar



ult. strength: rebar > splice

TENDONS: Ult. strength only slightly higher than yield stress. Also ductility low. They will break before rebars do, and before serious damage to anchorage zone.

REBARS (FOR R/C) OR TENDONS (P/C) WILL RUPTURE, MOST LIKELY AT UPPER PART OF CYLINDER OR AT DOME.

UNCERTAINTIES IN THE PROBLEM

LOAD: Magnitude of detonation
Initial conditions
Location of ignition
Etc.

RESISTANCE: Dynamic σ-ε & strength of steel rebars and splices
(bias, because of insufficient data)
Statistical variation of strengths.

RESULTS EXPECTED:

Detonation parameters
(magnitude, initial conditions, point of ignition)

↓

Time-histories of pressure on containment walls.

↓

Containment behavior: stress & strains in all components, assessment of damage (F.E.)

↓

Simple mechanical models

↓

Uncertainties (materials, load, etc.) → Simple probabilistic models. Assessment of safety.

SAND80-2501C

Eighth Water Reactor Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

Containment Safety Margins*

Walter A. Von Riesemann
Sandia National Laboratories
Systems Safety Technology Division

Introduction

The objective of the Containment Safety Margins program is the development and verification of methodologies which are capable of reliably predicting the ultimate load-carrying capability of light water reactor containment structures under accident and severe environments. Because the containment structure is a barrier to contain any fission products accidently released from within the reactor pressure boundary, it is important to know the margins of safety for containment structures under various extreme loading conditions.

The reliability of any predictive method should be verified through experiments. Since full scale testing is not feasible, models of containments will be used. Both overpressurization (static and dynamic) and earthquake loadings will be considered. As much as possible these loadings will be scenario independent.

The completion of this program will allow Licensing to make more reliable predictions of the capacities of containments and permit more realistic estimates of risk to public health and safety.

The program was initiated in June 1980 at Sandia and this paper will address the first phase of the program which is essentially a planning effort. Brief comments will be made about the second phase, which will involve testing of containment models.

Phase I - Planning

Efforts since June 1980 have been devoted to the establishment of a program plan which is to be completed by December 1980. Toward the attainment of this objective the following tasks are being pursued:

Task 1. Background Study - In this task, a literature survey to identify the existing analytical and experimental methods for the analysis of containments will be performed. Containment types will be categorized according to their structural design and behavior. Experimental capabilities of both the U.S. and other countries for seismic testing will be identified.

*This work is being performed by Sandia National Laboratories for the Structural Engineering Research Branch, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, under Memorandum of Understanding DOE 40-550-75.

Task 2. Evaluation and Development of Analysis Methods - Available numerical structural methods suitable for concrete and steel containments will be surveyed. The experimental data required to verif these numerical methods will be identified. If gross deficiences in the numerical methods are identified, state-of-the-art improvements will be pursued.

Task 3. Experimental Planning Effort - Because of the size and cost of containment structures, full scale tests are not feasible and only scale model tests will be conducted. An exception to this might occur if obsolete containments used for other than commercial power reactors are available for testing. This task will identify a test matrix which will include estimated costs and options.

An advisory panel of experts in containment structural analysis, design and behavior will be established to provide a broad-based input in establishing the experimental plan.

Results to Date

Since the program was begun in June 1980, Phase I has not been completed. The information given below includes some of the results which have been obtained.

Classification of Containments - The SAR°s (Safety Analysis Reports) for seventy operating plants were examined and the containments were classified according to type of construction. The results are given in the attached table. Also shown are cross-sections of some typical containment types.

Load Simulation - In performing the scale model tests, the various loading conditions will have to be simulated. The static overpressurization load offers no difficulty. For the dynamic overpressurization load either a combustible gas mixture, a point HE source, or an auxillary pressure tank with a rupture disk or similar technique will be used. At this time it is planned to do the axisymmetric loadings first and then, if necessary, asymmetric. Loadings with or without internal structural components will be performed.

To simulate an earthquake-like loading condition the following techniques are being investigated.

- underground explosives
- underground nuclear explosives (Nevada Test Site)
- shaker table (base excitation)
- moving mass exciters (above base excitation)
- chemical or mechanical pulsers
- pseudo-dynamic (calculated displacement applied to the structure)

Further study is required before any specific recommendations are made. A table listing some of the large shaker tables is attached.

Current Work

The current activities are directed toward completing Phase I (Planning) by the end of calendar year 1980. The major thrust of the planning document will be a recommended test matrix which will include such items as scale, testing method, analysis methods cost, expected results and options. At this time, we expect to conduct static overpressurization tests on free-standing steel containments first.

An advisory panel of people experienced in different aspects of containment design and behavior has been formed and they will provide input to the program plan.

Expected Program Results

As the experimental portion (Phase II) of the program progresses, knowledge of the load carrying capability of different types of containments subjected to overpressurization and earthquake loads will be established. This information will also be used to establish benchmark data for numerical analyses. With this information the NRC will be able to make more reliable predictions on capacities of containments.

Containment Safety Margins

- Work being performed by Sandia National Labs for Structural Engineering Research Branch Division of General Reactor Safety Research
- Initiated in June 1980

Objective

Development and verification of methodologies for reliably predicting ultimate load carrying capability of LWR containments under accident and severe environments

CONTAINMENT SAFETY MARGINS

LICENSING AND SAFETY ISSUE

- PROVIDE BASIS FOR STAFF DECISIONS FOR RELIABLE PREDICTIONS OF CONTAINMENT STRUCTURE CAPACITIES, WHICH WILL PERMIT MORE REALISTIC ESTIMATES OF RISK TO PUBLIC HEALTH AND SAFETY

Containment Safety Margins

Approach

- Combined analytical and experimental

  Phase I - Planning effort (to be completed by December 1980)

  Phase II - Experimental (multi-year effort)

Loadings

Scenario independent, where possible

- Overpressurization
  - Static
  - Dynamic (axisymmetric and asymmetric)
- Earthquake (lateral load capacity)

Containment Safety Margins

Phase I - Planning

Task 1. Background Study

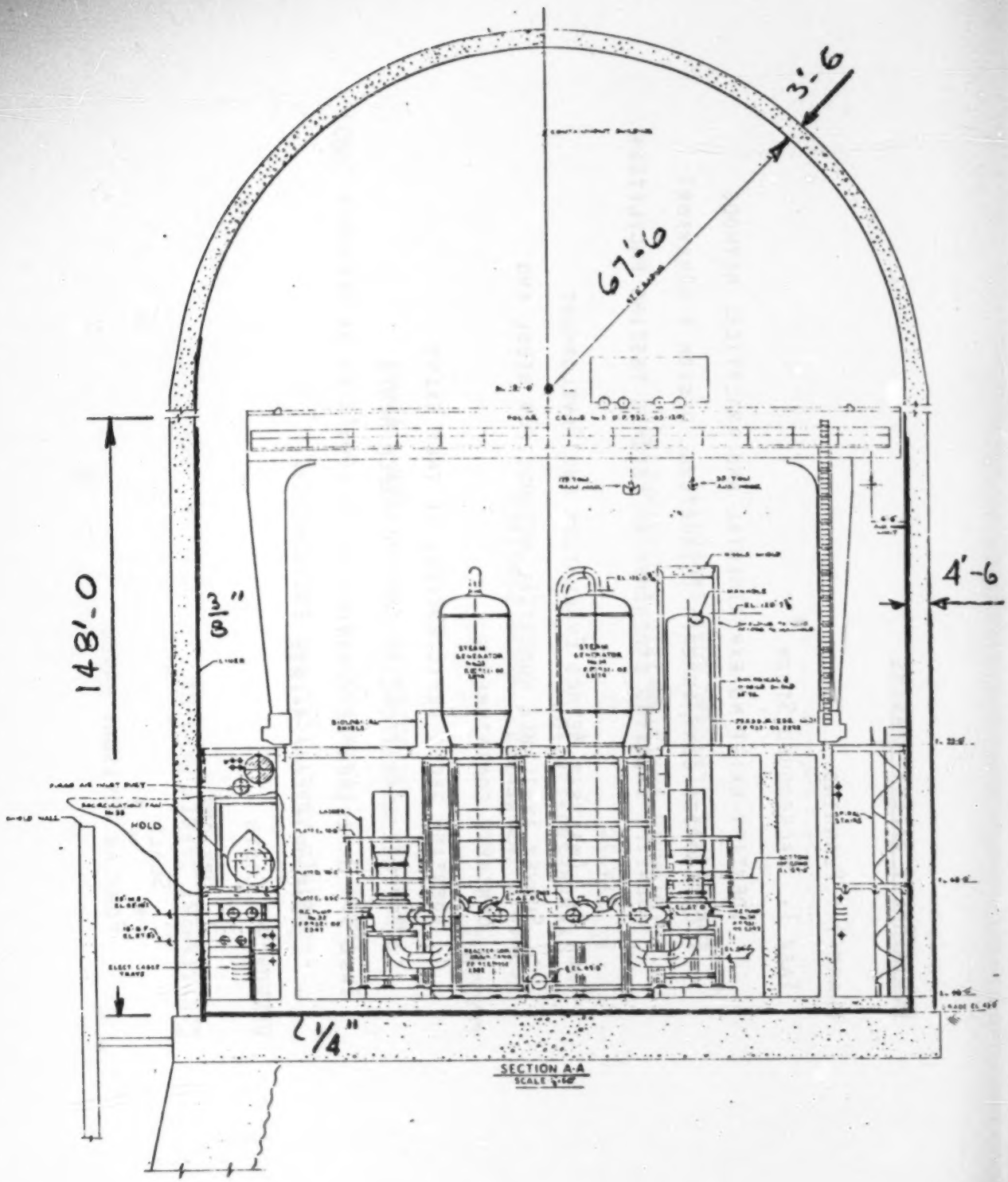
- Identify existing experimental and analytical methods
- Categorize containments (structural design & behavior)
- Identify existing experimental seismic testing facilities

Task 2. Analysis Method Evaluation and Development

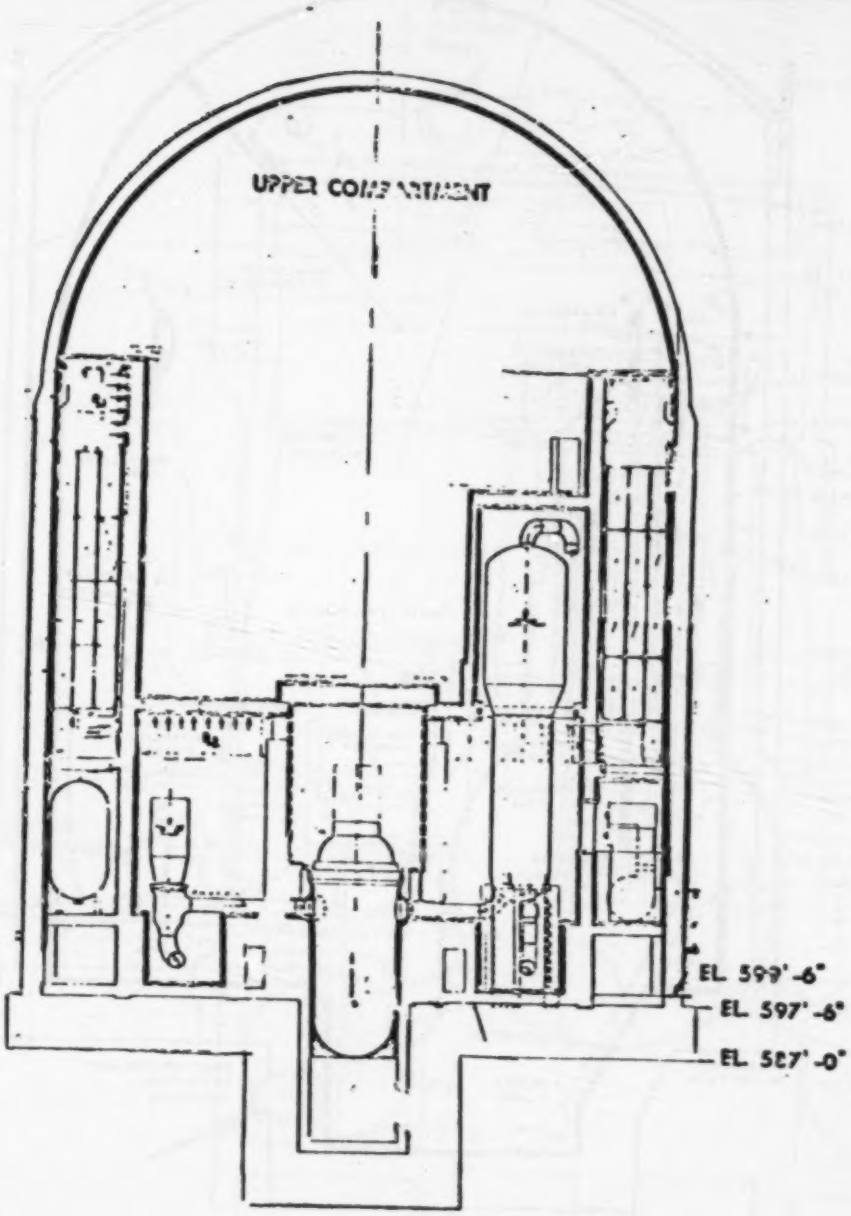
- Survey available numerical methods for steel and concrete containments
- Identify gross deficiencies, if they exist
- Develop analytical analogs to experiments

Task 3. Experimental Planning

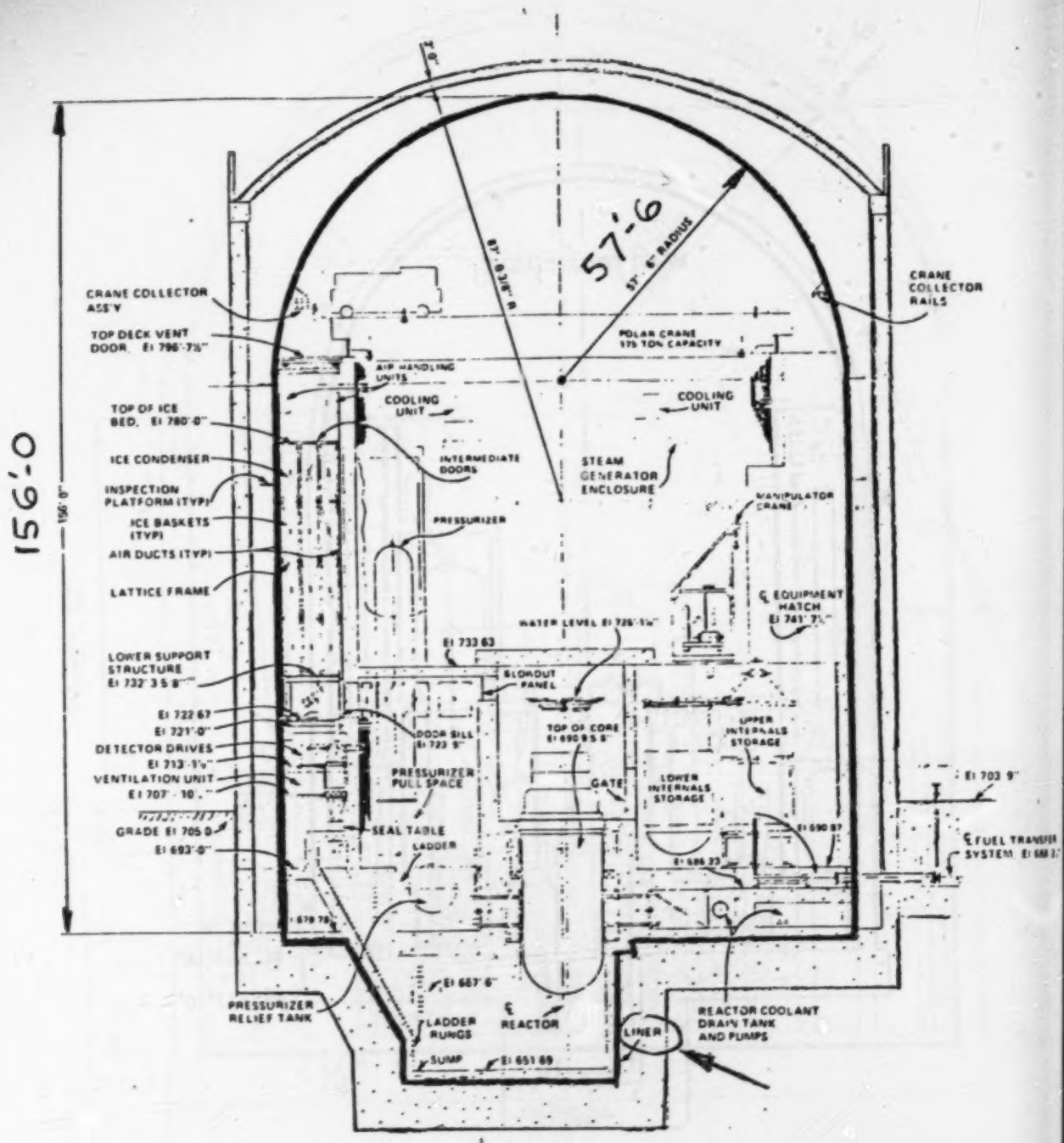
- Develop a test matrix
  - Costs
  - Options
  - Scale
- Form an advisory panel



INDIAN POINT 3



D. C. COOK
ICE-CONDENSER

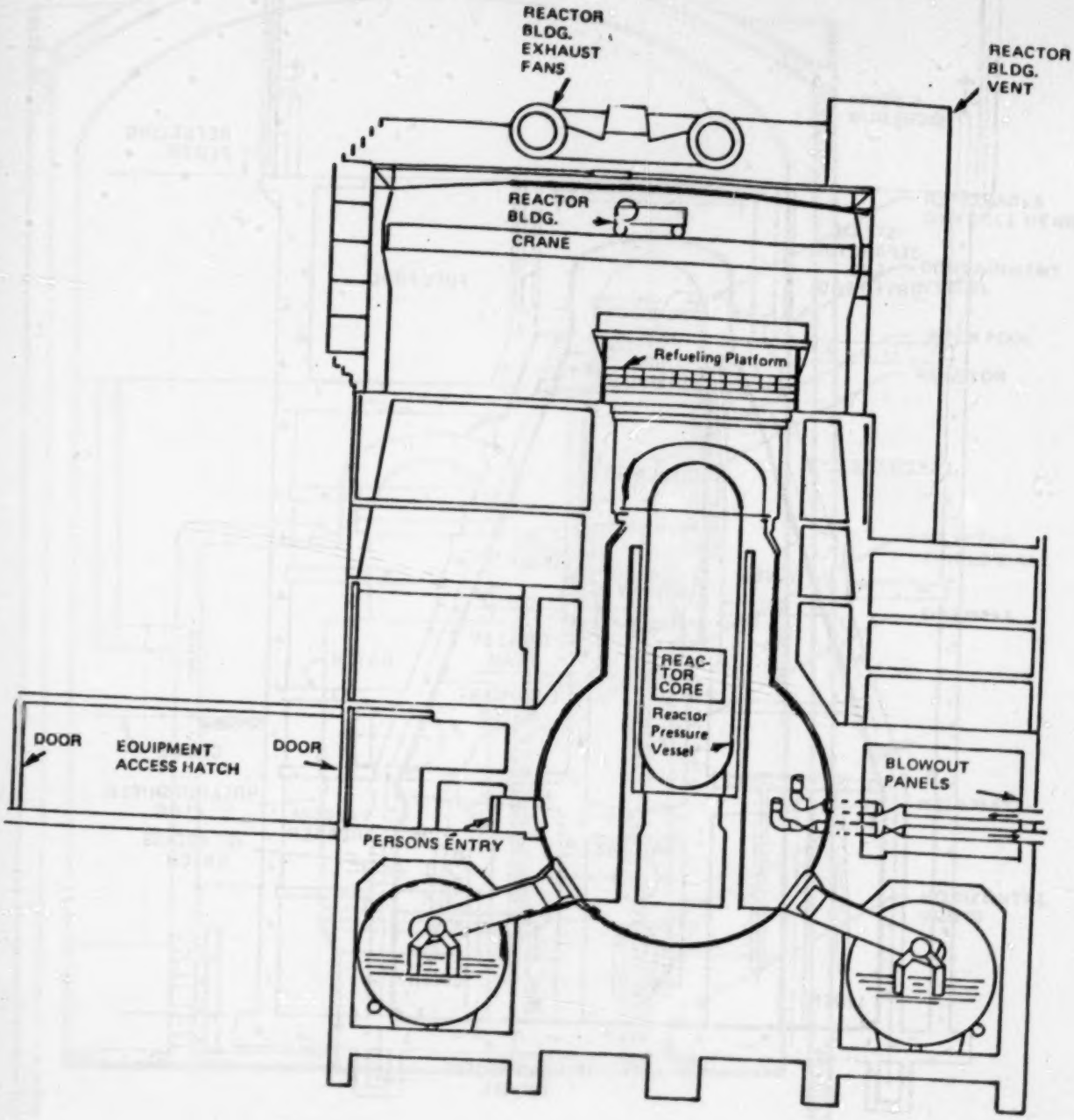


SECTION D12-D12

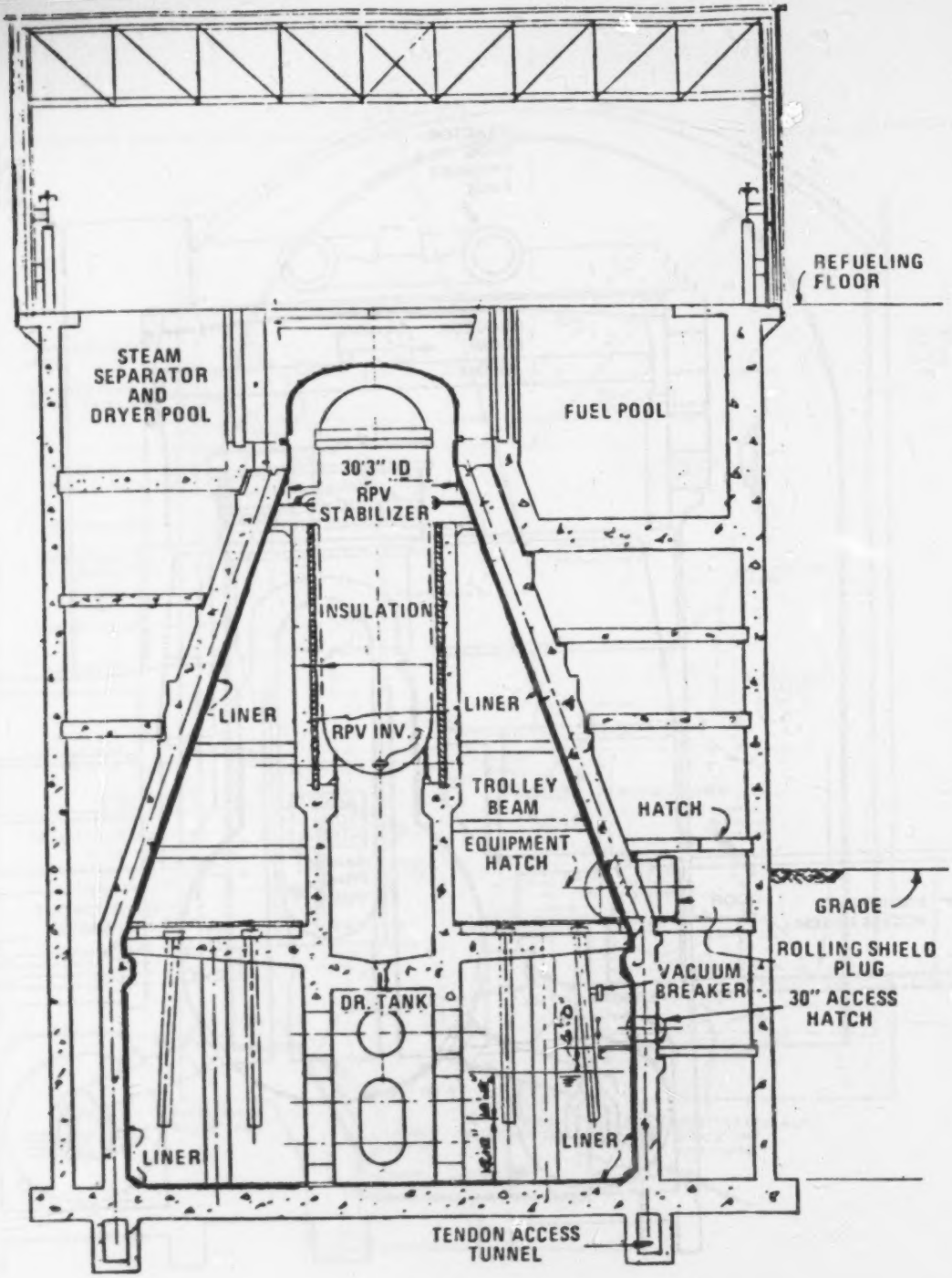
SEQUOYAH ICE-CONDENSER CONTAINMENT

TOTAL FREE VOLUME 1.2×10^6 FT^3

DESIGN INTERNAL PRESSURE 10.8 PSIG

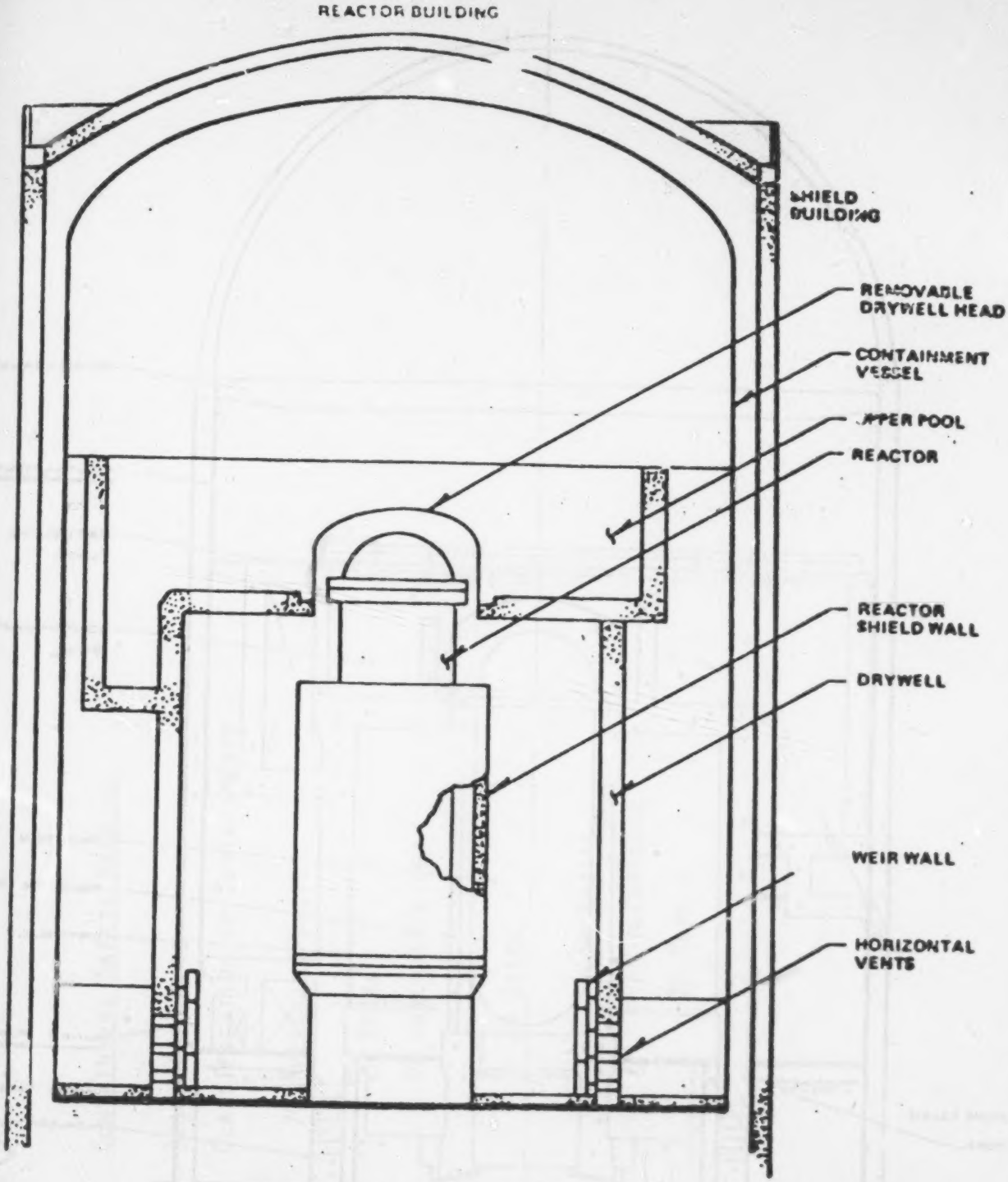


MARK I CONTAINMENT



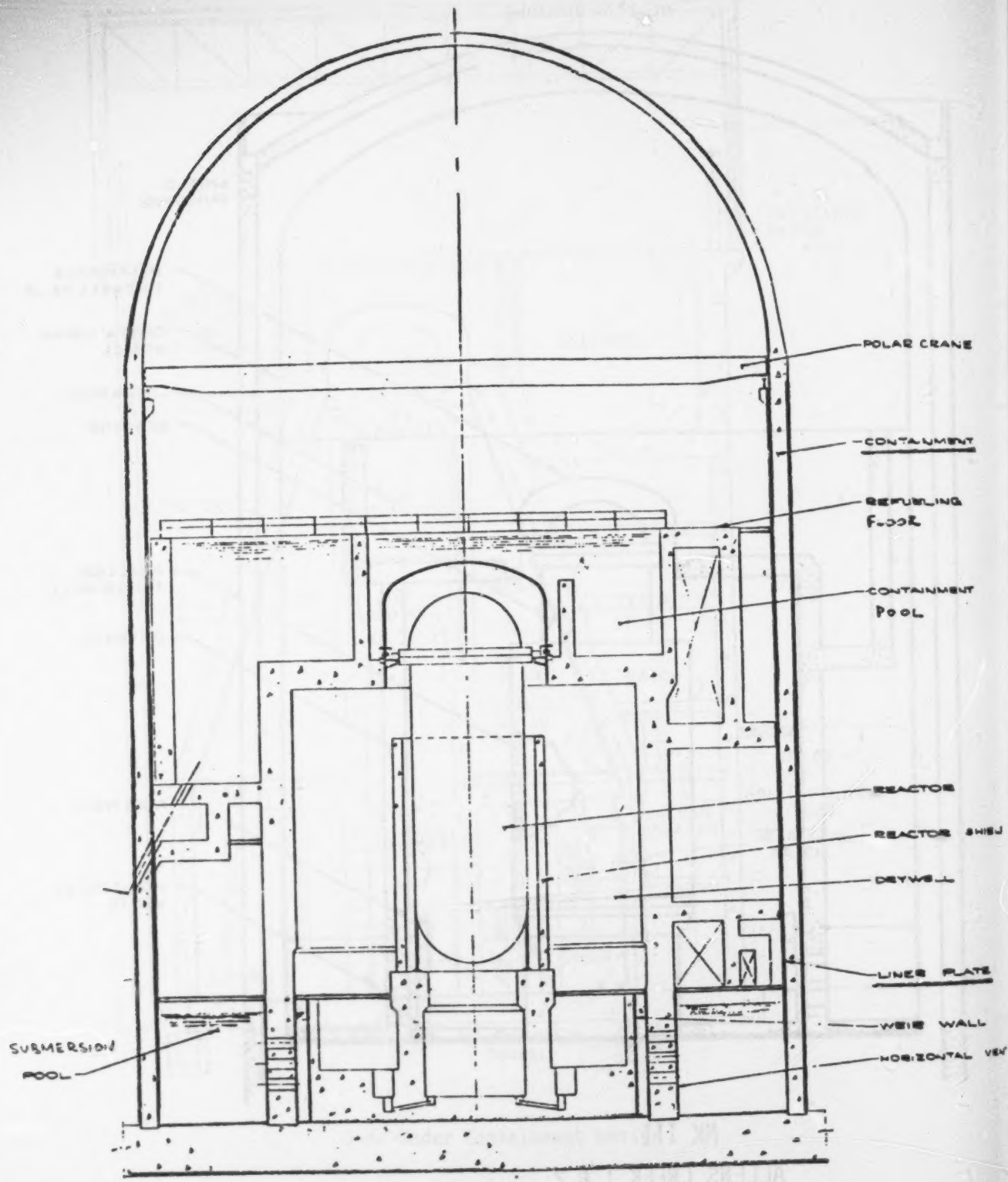
Over-Under Containment Design

MARK II CONTAINMENT



MK III

ALLENS CREEK 1 & 2



MK III

SKAGIT

Containment Safety Margins

Results to Date

Classification of Containments

- 70 Operating Units
  - 29 Steel
  - 41 Concrete/Liner
    - 29 Steel
      - 4 Spherical
      - 5 Cylindrical
      - 20 Mark I

Containment Safety Margins

Classification of Components (Cont'd)

- 41 Concrete
  - 16 Deformed Bar
    - 5 Suppression
      - 3 Mark I
      - 2 Ice-Condensers
    - 6 Sub-Atmospheric
    - 5 Atmospheric
  - 4 Mixed
  - 21 Prestressed
    - 15 Six Buttress, Shallow Dome
    - 5 Three Buttress, Shallow Dome
    - 1 Three Buttress, Hemispherical Dome

"Design" Pressure 10.8 to 47 psig

Free Volume 1.2 to 2.6 x 10^6 FT^3.

Containment Safety Margins

Results to Date

Load Simulation

Load Simulation

- Overpressurization
  - Static
  - Dynamic
    - Combustible Gas Mixture
    - HE
    - Pressure Tank/Rupture Disk
- Earthquake
  - Underground high explosives
  - Underground Nuclear Tests (Nevada Test Site)
  - Shaker Table (Base Excitation)
  - Moving Mass Exciters
  - Pulsers (Rocket Motors, Steam, Cutting Machine)
  - Pseudo-Dynamic (Static Displacement of Structure)

CONTAINMENT SAFETY MARGINS

RESULTS TO DATE

SURVEY OF SHAKER TABLES

LOCATION	FORCE RATING 10^6 LB	TEST AREA (FT)	COMMENTS
NUPEC, JAPAN	H 6.6 V 7.2	49x49	FOR 1.1×10^6 LB TEST WEIGHT MAX. ACCEL H 2.6 G V 1.3 G MAX. TEST WEIGHT 2.2×10^6 LB
CERL, ILLINOIS	H .45 V .81	12x12	13,000
OGDEN, UTAH (HILL AFB)	.2	262 FT^2	87,000
UNIV. OF CALIFORNIA (RICHMOND)	.1	20x20	30,000
WYLIE LABS	.15	12x12	40,000

MAX. 2 G TEST ITEM WT. (LB)

NOTE - NUPEC IS PLANNING TO TEST A 1/4 SCALE PWR CONTAINMENT ($.66 \times 10^6$ LB) AND A 1/3 SCALE BWR CONTAINMENT ($.77 \times 10^6$ LB)

Containment Safety Margins

Current Work

- Program Plan completed by January 1981

  Test Matrix

  - Size
  - Test Method
  - Cost
  - Options

- Advisory Panel has been formed

CONTAINMENT SAFETY MARGINS

EXPECTED RESULTS

PHASE II - EXPERIMENTAL EFFORT

- LOAD CARRYING CAPABILITY OF CONTAINMENTS SUBJECTED TO OVERPRESSURIZATION AND EARTHQUAKES ESTABLISHED
- VERIFED NUMERICAL CODES
- BENCH MARK DATA FOR NUMERICAL ANALYSES FOR DIFFERENT TYPES OF CONTAINMENTS

Evaluation of Dynamic Testing of Nuclear Power Plant Structures

M. G. Srinivasan, C. A. Kot, B. J. Hsieh and H. H. Chung

Components Technology Division
Argonne National Laboratory
Argonne, Illinois 60439

Presented at the
U.S. NRC
8th Water Reactor Safety Research Information
Meeting

Gaithersburg, Maryland
October 27-31, 1980

EVALUATION OF DYNAMIC TESTING OF NUCLEAR POWER PLANT (NPP) STRUCTURES

SUMMARY

Current practice both for design and licensing of NPP structures relies entirely on estimating structural response by means of linear analytical models. Because of the complexity of the structures and other uncertainties involved, the analytical models are usually based on assumptions intended to make the design conservative. Whereas dynamic testing is often needed for the qualification of certain NPP equipment, such tests are not required for structures.

The objective of the current program is to provide a rational basis for decisions concerning the inclusion of dynamic testing of NPP structures as part of the licensing process. This is to be accomplished through an integrated evaluation of dynamic testing methods and associated analytical procedures. The following concerns specifically the interim evaluation of dynamic tests of as-built NPP structures, which have been performed over the past 20 years both in the U.S.A. and abroad. The purpose of most of these experiments was to validate analytical and computational methods used in the design. Only in one or two cases were the tests used to improve the analytical model and to predict the seismic response.

A variety of excitation techniques at various levels of excitation (represented by the peak response) have been employed in the experiments. However, in most cases the excitation levels are only about one hundredth of natural earthquake levels. Specific techniques used are: ambient loading, steady state vibration by means of mechanical, hydraulic or electrodynamic shakers, transient or pulse train loading by means of pulse generators, rockets or buried explosives. Free vibrations arising out of snapback are used mostly on equipment.

Each of these techniques has certain advantages and shortcomings. Thus ambient testing has the advantage of comparatively minimal testing effort. Yet the very low levels of excitation limits its usefulness in response simulation for high level excitations. Forced vibration using rotating mass vibrations is the most widely adopted technique in the tests. It provides information on natural frequencies and permits mode isolation. Though this method is capable of generating high response levels, it is difficult to apply for systems with closely spaced modes. Pulse generators and rockets are directly attached to the structure and they excite the structure by impulsive or transient loading. Using these methods it is possible, in principle, to apply a pulse train to the structure to simulate actual earthquake response (though at lower levels). In all the above mentioned methods the source of excitation is directly applied on the structure and therefore the extent of soil-structure interaction is probably not the same or even similar to that in an actual earthquake. The method of excitation by means of detonating buried explosives is therefore better than the above methods in that context. In this method an array of buried explosives whose location, spacing, firing sequence etc. are calculated so as to simulate earthquake-like

ground motion are fired to produce high level response. Though this method is capable of generating design earthquake level response other safety considerations limit the maximum excitation levels to lower values. In all tests modal parameters are determined employing Fast Fourier Transforms (FFT) or similar techniques. In a few cases the experimental results were used to re-calculate parameters of the linear analytical model.

Certain general trends emerge from the past experiments. Tests in which different levels of excitation were used showed that the modal parameters varied with excitation levels. Though such non-linear behavior is noted it has not been possible to quantitatively characterize the nonlinearity. The assumed viscous damping appears to be largely influenced by soil-structure interaction and very much site dependent. In the case in which actual earthquake response records were compared with response predicted on the basis of a linear analytical model with experimentally determined parameters, response levels differed by a factor of three. In tests employing more than one excitation technique, it was found that though there is generally good agreement in terms of determined dynamic parameters instances of one technique failing to indicate a mode shown to occur by another techrique do arise.

The following is a preliminary evaluation of dynamic tests on nuclear power plant structures. Whereas the ultimate purpose is to have the assurance that the structural response of a structure to a design dynamic loading will be within permissible (design) limits, dynamic tests could not be used to provide this assurance directly since proof testing of an actual structure is impractical. Only a combination of low level testing and analytical techniques for extrapolation of low level response to design level response is practical. Past experience shows that it is feasible to test structures at almost any desired level of excitation, to measure and process the signal and to analyze the data using sophisticated digital techniques. With regards to the associated analytical methodologies the picture is less clear. Though various analytical techniques have been proposed for applying the data towards generating a reliable model, none of them have been validated using actual recorded data. It must, however, be pointed out that the above conclusions are based largely on nuclear power plant tests. A large amount of literature on dynamic tests and related analysis on non-nuclear structures exist and remains to be reviewed. Other problems which must be resolved in the course of this study are the repeatability of tests on as-built structures, the role of scale model tests for extrapolation purposes and as adjuncts to tests on the NPP structures, and the performance and design adaptability of various analytical techniques in response prediction. Finally the most promising analytical techniques will be subjected to a validation procedure. This will lead to a final evaluation of the role of dynamic testing of nuclear power plant structures in the licensing procedure.

EVALUATION OF DYNAMIC TESTING METHODS FOR SAFETY ASSESSMENT OF NUCLEAR POWER PLANT (NPP) STRUCTURES

PROGRAM STARTING DATE - MARCH 1980

BACKGROUND

- CURRENT DESIGN AND LICENSING OF NPP STRUCTURES IN GENERAL BASED ON LINEAR ANALYSIS PROCEDURES

- DIVERSE OPINIONS EXIST CONCERNING PROPER METHODS FOR SPECIFIC CASES, RANGE OF VALIDITY OF PARTICULAR METHODS, AND RELATIONSHIPS OF ANALYSES TO EXPERIMENTAL OBSERVATIONS

- UNCERTAINTY LEADS TO CONSERVATISM WHENEVER QUESTIONS OF APPLICABILITY ARE RAISED

MOTIVATION FOR EVALUATING DYNAMIC TESTING OF NPP STRUCTURES

- ESTABLISH RATIONAL CONNECTION BETWEEN RESPONSE PREDICTED BY ANALYSIS AND ACTUAL RESPONSE OF AS-BUILT STRUCTURES PARTICULARLY UNDER SEVERE LOADING CONDITIONS

- REDUCE UNNEEDED CONSERVATISM IN DESIGN METHODS USED IN LICENSING PROCESS

- VERIFY AND VALIDATE DESIGN METHODOLOGIES AND PREDICTION PROCEDURES

- PRECEDENT - QUALIFICATION AND OTHER TESTING OF NPP EQUIPMENT AS PART OF LICENSING PROCESS

OBJECTIVE

- PROVIDE A RATIONAL BASIS FOR POSSIBLE INCLUSION OF DYNAMIC TESTING OF NPP STRUCTURES AS PART OF THE LICENSING PROCESS
- EVALUATE DYNAMIC TESTING METHODS
- EVALUATE ANALYTICAL MODELS WHICH USE EXPERIMENTAL RESULTS
- VERIFY AND VALIDATE COMBINED METHODOLOGIES
  - AS-BUILT UNDAMAGED STRUCTURES
  - DAMAGED STRUCTURES

TASK STRUCTURE

TASK 1: FEASIBILITY, EVALUATION AND UTILIZATION OF DYNAMIC TESTS ON AS-BUILT STRUCTURES

TASK 2: EVALUATION AND VERIFICATION OF ANALYTICAL METHODS UTILIZING DYNAMIC TEST DATA

TASK 3: EVALUATION OF DYNAMIC TESTING FOR THE ASSESSMENT OF DAMAGE AND THE REQUALIFICATION OF DAMAGED STRUCTURES

TASK 1

EVALUATION AND UTILIZATION OF DYNAMIC TESTS ON EXISTING STRUCTURES

- PARTICIPATION IN HDR TEST PROGRAM
- EVALUATION OF NPP STRUCTURAL EXPERIMENTAL DATA
- REVIEW AND EVALUATION OF NON POWER PLANT STRUCTURAL TESTING
- INTERIM EVALUATION OF DYNAMIC TESTING FEASIBILITY

TASK 2

EVALUATION AND VERIFICATION OF ANALYTICAL METHODS UTILIZING DYNAMIC TEST DATA

- EVALUATION OF SYSTEMS IDENTIFICATION AND OTHER PERTINENT METHODOLOGIES
- VALIDATION OF SPECIFIC METHODOLOGIES
- RELATIONSHIP OF METHODOLOGIES TO DESIGN PRACTICE

TASK 3

EVALUATION OF DYNAMIC TESTING FOR THE ASSESSMENT OF SEISMIC AND POST-ACCIDENT DAMAGE AND THE REQUALIFICATION OF DAMAGED STRUCTURES

- DEFINITION OF DAMAGE AND EXTENSION OF SYSTEM IDENTIFICATION TO INCLUDE DAMAGE ESTIMATION

- RECOMMENDED DYNAMIC TESTS AND STRUCTURAL REQUALIFICATION

CURRENT PROGRAM STATUS

- MAJOR EMPHASIS ON TASK 1
- CRITICAL REVIEW OF DYNAMIC TESTS ON NPP AND ASSOCIATED ANALYSIS
- STARTED THE EVALUATION OF PAST AND CURRENT HDR TESTS
- STARTED THE ASSESSMENT OF EXPERIMENTS ON STRUCTURES OTHER THAN NUCLEAR POWER PLANTS
- INTERIM EVALUATION OF FEASIBILITY OF DYNAMIC TESTING

DYNAMIC TESTS ON NUCLEAR POWER PLANT STRUCTURES

- U.S.A.: UCLA RR, CVTR, EGCR, FERMI, SONGS, HUMBOLDT BAY
- JAPAN: NUC. SAFETY RR, FUKUSHIMA, SHIMANE, HAMAOKA NO. 1 & 2
- ITALY: PEC (FAST REACTOR FOR MAT. TEST.), CAORSO
- GERMANY: HDR
- U.K.: DUNGENESS "B"

GOALS OF PAST DYNAMIC TESTS

A. DETERMINATION OF DYNAMIC CHARACTERISTICS

1. VALIDATION OF DESIGN METHODS
2. IMPROVEMENT OF PREDICTIVE ANALYTICAL MODEL

B. RESPONSE SIMULATION

1. AT HIGH LEVELS OF EXCITATION:
   EARTHQUAKE SIMULATION
2. AT LOW LEVELS OF EXCITATION

ANALYTICAL MODELS

DESIGN/PRE-TEST/A-PRIORI

- TYPES: 2D, AXI-SYMMETRIC, 3-D FEM, 3D-LUMPED MASS-SPRING
- LINEAR: $M\ddot{X} + C\dot{X} + KX = F(t)$
- DAMPING: VISCOUS, STRUCTURAL
- SOLUTIONS: EIGENVALUES, EIGENVECTORS, MODAL DAMPING, RESPONSE CURVES

EXCITATION LEVELS

- FEASIBLE PEAK RESPONSE LEVELS: 10^{-4} g - 1.0 g
- RELATION TO GOAL - UNFEASIBILITY OF PROOF TESTS
- ASSUMPTION OF LINEARITY
- SAFETY CONSIDERATIONS

TESTING METHODS: EXCITATION TECHNIQUES

- AMBIENT LOADING
- VIBRATORS: ROTATING ECCENTRIC MASS, HYDRAULIC, ELECTRODYNAMIC
- PULSE GENERATORS
- ROCKETS
- BURIED EXPLOSIVES
- SNAPBACK (FREE VIBRATIONS)

AMBIENT LOADING

- WIND, TRAFFIC, MICROTREMORS
- PEAK RESPONSE LEVEL: 10^{-4} TO 10^{-3} g
- ADVANTAGES & DISADVANTAGES
- TESTING EXPERIENCE

ROTATING ECCENTRIC MASS VIBRATOR

- STEADY STATE SINUSOIDAL FORCE (UNI OR OMNI DIRECTIONAL)
- PEAK RESPONSE LEVELS: UP TO 1g
- SINESWEEP, SINE DWELL, MODAL TUNING
- ADVANTAGES & DISADVANTAGES
- TESTING EXPERIENCE

READEX FICHE best copy available

BURIED EXPLOSIVES

- SIMULATION OF EARTHQUAKE-LIKE GROUND MOTION, IMPULSE
- PEAK RESPONSE LEVEL: UP TO 1g
- PRE-TEST ANALYSIS FOR LOCATION, SPACING, TIME DEALY
- ADVANTAGES & DISADVANTAGES
- TESTING EXPERIENCE

PULSE GENERATORS

- PULSE TRAINS: OPTIMAL SIMULATION OF SEISMIC RESPONSE
- PEAK RESPONSE LEVEL: $\sim 10^{-2}$ g
- PRE-TEST ANALYSIS FOR DESIGN OF PULSE TRAIN
- ADVANTAGES & DISADVANTAGES
- TESTING EXPERIENCE

ROCKET/ROCKET MOTOR

- IMPULSE FROM REACTION TO HIGH VELOCITY EJECTION
- PEAK RESPONSE LEVEL: $\sim 10^{-2}$ g
- ADVANTAGES & DISADVANTAGES
- TESTING EXPERIENCE

DETERMINATION OF DYNAMIC CHARACTERISTICS

- MODAL PARAMETERS
- FOURIER REPONSE SPECTRUM
- POWER SPECTRAL DENSITY FUNCTION
- FREQUENCIES: RESONANCE PEAKS
- DAMPING: HALF-POWER BAND WIDTH OR LOG. DECREMENT
- MODE SHAPES: MODAL TUNING
  FREQUENCY RESPONSE FUNCTION

GENERAL TRENDS FROM TEST RESULTS

- DEPENDENCE ON EXCITATION TECHNIQUES
- DAMPING
- NON-LINEAR BEHAVIOR

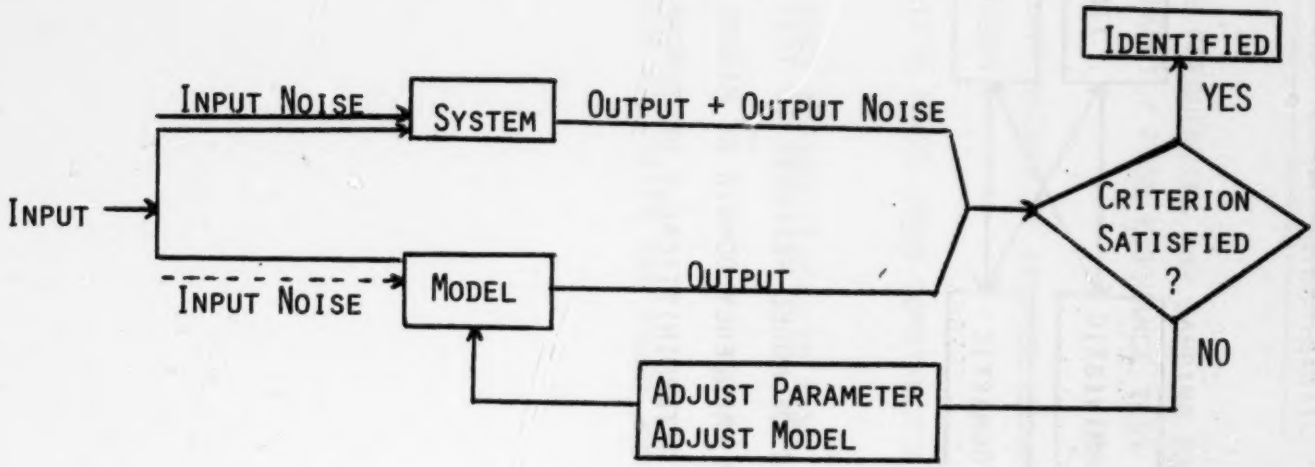
POST-TEST ANALYTICAL MODELING

1. IMPROVEMENT OF A-PRIORI MODEL USING TEST RESULTS

   - DIRECT PARAMETER SEARCH (EGCR)
   - PARAMETER ID TECHNIQUE (SONGS)

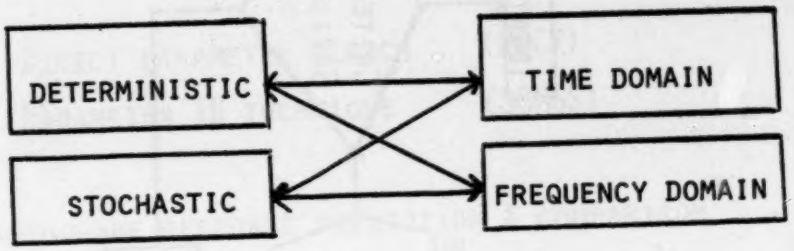
2. EARTHQUAKE RESPONSE PREDICTION & COMPARISON

SYSTEM IDENTIFICATION FLOW-CHART



PARAMETER ESTIMATION

METHODS:



BEST FIT OF MODEL RESPONSE TO TEST RESPONSE IN TIME OR FREQUENCY DOMAIN BY MINIMIZING A CRITERION EITHER DETERMINISTICALLY OR STOCHASTICALLY.

DIRECT SEARCH METHOD-DETERMINISTIC

- APPROXIMATE FREQUENCIES AND DAMPING RATIOS OBTAINED DIRECTLY FROM RESONANCE TEST
- MODAL COUPLING MATRIX IS ASSUMED KNOWN
- SIMPLE COMPLETE MODEL THEN CONSTRUCTED
- TRIAL AND ERROR METHOD IS USED TO FIND BEST VALUES
- NEED KNOWLEDGE OF SYSTEM AND SOUND JUDGMENT

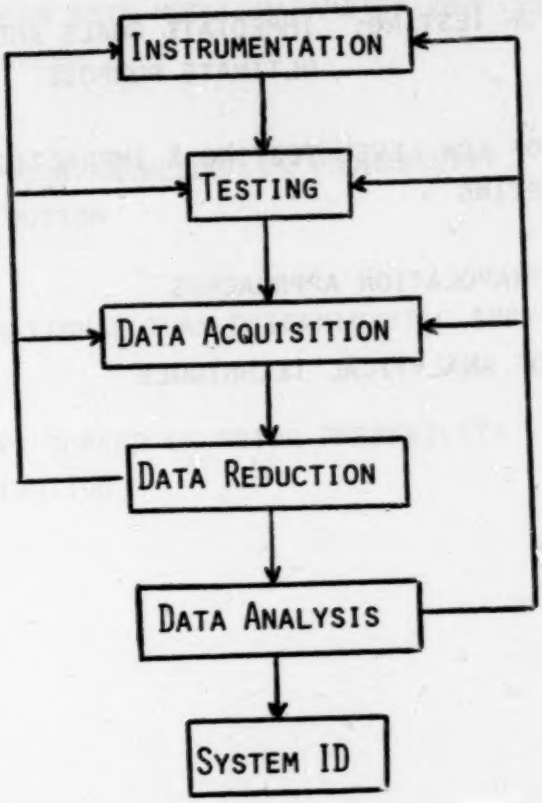
BAYESIAN METHOD - STOCHASTIC

- CONSIDERS BOTH MODEL PARAMETER AND TEST DATA RANDOM
- NEED PRIOR KNOWLEDGE ON PROBABILITY DISTRIBUTION
- MORE RATIONAL THAN DETERMINISTIC APPROACH
- RESULTS DEPEND ON PRIOR PROBABILITY DISTRIBUTION

EVALUATION OF DYNAMIC TESTING

- PHILOSOPHY OF TESTING: IMMEDIATE GOALS AND ULTIMATE PURPOSE
- INADEQUACY OF LOW LEVEL TESTING & IMPRACTICALITY OF PROOF TESTING
- PROPOSED EXTRAPOLATION APPROACHES
- INADEQUACY OF ANALYTICAL TECHNIQUES

TESTING AND ANALYSIS PROCEDURE



PROBLEM AREAS

- NOISE
- CLOSELY-SPACED RESONANCES
- NONLINEARITY

PROBLEMS OF CONVENTIONAL SPECTRUM ANALYSIS

- ALIASING
- LEAKAGE
- STATISTICAL INSTABILITY
- BIAS ERROR
- FREQUENCY RESOLUTION LIMITED BY WINDOW WIDTH
- POOR FOR SHORT SIGNAL & CLOSELY-SPACED RESONANCES

MODELING APPROACH TO SPECTRAL ANALYSIS

- AUTOREGRESSIVE MODEL
- MAXIMUM ENTROPY SPECTRAL ANALYSIS
- EXPONENTIAL MODELS
- ADVANTAGES
  - HIGHER RESOLUTION
  - GOOD FOR SHORT SIGNALS AND CLOSELY-SPACED RESONANCES
  - NO LEAKAGE
  - STATISTICAL STABILITY

NON-PARAMETRIC SYSTEM IDENIFICATION

- CAN BE APPLIED FOR NON-LINEAR SYSTEM

- RESTORING FORCE REPRESENTATION

$$F_R = F_R\ (X,\ \dot{X})$$

ISSUES FOR FURTHER INVESTIGATION

- TESTING EXPERIENCE WITH NON-NPP STRUCTURES
- REPEATABILITY OF TESTS
- PROPOSED A-PRIORI & POST-TEST ANALYTICAL TECHNIQUES
- ROLE OF SCALE MODEL TESTS

CODES AND STANDARDS WITH RELATION TO CONTAINMENT SAFETY MARGINS

by Richard N. White and Peter Gergely, Cornell University

INTRODUCTION

Three topics are treated in this paper. The first two are on research in progress and the third is on a related topic that needs research. The topics are:

1. Peripheral (punching) shear in reinforced concrete walls subjected to simultaneous biaxial tension.
2. Tangential (membrane) shear in reinforced concrete walls subjected to simultaneous biaxial tension, with special reference to containment structures.
3. Radial shear in reinforced concrete containment walls.

PERIPHERAL SHEAR

This loading condition occurs in containments and other safety-related nuclear structures at attachments to the walls and at penetrations through the walls. The current design criterion from Article CC-3421.6 of Section III, Division 2 of ASME is given in Plate 2, where shear strength v_u of the concrete is plotted against biaxial tension level in the reinforced concrete. It is seen that with this criterion, shear strength decreases very rapidly with increasing biaxial tension. This leads to the requirement of very substantial amounts of reinforcing steel to carry peripheral shear, which in turn creates severe congestion problems and difficulties in placing high quality concrete.

26 experiments were conducted on flat reinforced concrete slabs to determine the influence of biaxial tension on punching shear strength. Geometry, design, materials, test conditions, loading, and behavior are summarized in Plates 3-8, and results are plotted in Plate 9. All specimens were precracked in biaxial tension prior to the application of shear load. Biaxial tension levels ranging from zero to 80% of the yield strength of the reinforcement were maintained during the shear load. Eqns. 1b and 1c in Plate 9 are for shear measured on a perimeter located at d/2 from the face of the load plate, and Eqn. 1a is for shear measured along the actual failure surface which coincided with the orthogonal crack patterns induced by the biaxial tension (crack spacing of about 6 inches).

A second independent test series with much larger shear spans is reported by Johnson (Plates 10-12). His results, which included specimens with biaxial tensions of 0, $0.43f_y$ and $0.86f_y$, plot even higher than the Cornell results. A correlation with Mattock's expression for shear strength in the presence of uniaxial tension is given in Plate 13.

The conclusions listed in Plate 14 indicate the excessive conservatism involved in the present criteria, and the need for the development of a fully rational criteria for peripheral shear design. Planned research will permit the systematic investigation of the critical parameters involved, but all indications point to the fact that reinforced concrete under high levels of biaxial tension still has appreciable peripheral shear strength.

TANGENTIAL SHEAR

A critical design loading for reinforced concrete containments is internal pressurization combined with seismic shear force. The internal pressurization may produce extensive cracking in the containment wall, across which large seismic shear forces must be transmitted. ASME Section III, Division 2 provisions permit modest amounts of shear force to be carried by the concrete in conjunction with a two-way reinforcement pattern, that is, hoop bars and vertical bars in the containment wall. These provisions are based on tests carried out at Cornell University about 10 years ago on uniaxially tensioned specimens that were greatly simplified in terms of actual behavior in the containment.

Current design methods require the use of four-way reinforcement (diagonal steel in two directions in addition to the orthogonal vertical and horizontal steel) to carry seismic shear in excess of that permitted by ASME CC-3421.5. This special reinforcement, all curved, produces considerable congestion in the containment and hinders construction.

Hence the objective of this phase of the research program was to gain a better understanding of the mechanisms of in-plane shear transfer across cracks in biaxially tensioned reinforced concrete, and to use this understanding to formulate a fully rational design and licensing criteria for reinforced concrete containment structures. The research results also have applicability to other nuclear structure walls that are loaded in both tension and shear.

The research approach is summarized in Plate 16. Flat reinforced concrete elements were used to simulate a section of the cracked containment wall (Plate 17); the specimen and the method of introducing reversing shear forces into the biaxially tensioned concrete are both illustrated in Plate 19. Shear forces were introduced through thickened corner regions, with all instrumentation concentrated in the thinner central region. Specimen configuration and parameters, as well as types of loading history used, are summarized in Plates 20 and 21. Nearly 50 specimens have been tested to date.

Typical cracking patterns produced by the biaxial tension and the reversing shear forces are plotted in Plate 22, and typical shear stress-shear strain response is given in Plate 23 for a specimen carrying $0.6f_y$ biaxial tension in the reinforcement during shear loading. 10 complete shear reversals were applied at each of the several levels of shear (125, 175, 225 psi) and failure occurred during the second cycle at 250 psi. Shear stiffnesses for monotonically and cyclically sheared specimens are given in Plates 24 and 25. A representative value of shear stiffness of the cracked specimens is about 5% of that of uncracked concrete, with values that decrease with increasing biaxial tension. Cycling further degrades stiffness, with extremely low stiffnesses at shear stresses below about 25 psi (because the slightly open cracks have not yet come into contact). Slopes of idealized shear stress-shear strain relationships, given in terms of the fraction of shear stiffness G_o for uncracked concrete, are given in Plate 25 for various combinations. A similar idealization with cyclic degradation has been incorporated into a dynamic analysis capability for containment structures.

Failure occurred in all cases by yielding of reinforcing steel. Ultimate shear strength v_u is plotted against a reinforcing parameter in Plate 26, with a conservative lower bound expression given on the plot. Japanese tests on model containment-type structures (Plate 27) are compared with Cornell results in Plate 28, along with results from the PCA program on large-scale specimens.

Conclusions are given in Plates 29 and 30. Strength is quite predictable from a simple equilibrium model that also includes the dowel action contributions of the steel. Stiffness is surprisingly low and the implications of this low shear stiffness on strains and stresses in the liner need extensive assessment (see Plate 30). At this point in time in the research program it is believed that two-way reinforced containments can carry substantial seismic shear even with high biaxial tension, but that the limit on this type of design may well be deformation-controlled.

RADIAL SHEAR: Radial shear at or near the base of a containment is produced by internal pressurization. Current design criteria are not based on experimental evidence and are most likely extremely conservative. The research on punching and tangential shear, combined with new research that includes the effects of hoop steel, role of dowel forces, influence of partial crack closing, etc., will lead to improved design criteria for radial shear.

REFERENCES

1. "Punching Strength of Biaxially Tensioned Reinforced Concrete" by R.N. White, J. Abrams, and P. Gergely, Proc. of ASCE Conference on Civil Engineering and Nuclear Power, Vol. 1, Materials and Structural Design, Knoxville, September 1980, 15 pp.
2. "Tangential Shear Design Considerations" by C. Conley, P. Perdikaris, R. White, and P. Gergely, same conference and volume as Ref. 1, 28 pp.
3. Strength and Stiffness of Tensioned Reinforced Concrete Panels Subjected to Membrane Shear, Two-Way Reinforcing, by P. Perdikaris, R.N. White, and P. Gergely, USNRC Report NUREG/CR-1602, 394 pp, Aug. 1980.
4. "Strength and Stiffness of Reinforced Concrete Containments Subjected to Seismic Loading" by R.N. White, P. Perdikaris, and P. Gergely, Nuclear Engineering and Design, Vol. 52/1, (Also published in Trans. of 5th SMiRT Conference, Berlin, Panel Session JP, pp. 77-113).
5. Strength and Stiffness of Tensioned Reinforced Concrete Panels Subjected to Membrane Shear, Two-Way and Four-Way Reinforcing, by C. Conley, R.N. White, and P. Gergely, to be published in fall 1980 as USNRC NUREG Report, 165 pp.

PUNCHING STRENGTH

OF

BIAXIALLY TENSIONED CONCRETE

SPONSOR: NRC

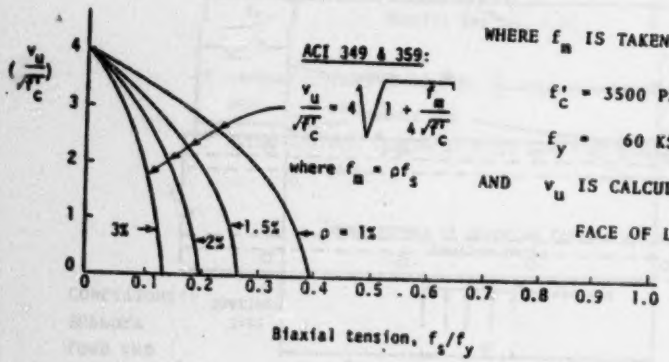
RESEARCH OBJECTIVE

TO DETERMINE THE DEPENDENCE OF SLAB PUNCHING STRENGTH ON THE LEVEL OF TENSION APPLIED TO ORTHOGONAL BARS IN SLAB.

(1)

CURRENT DESIGN CRITERIA:

FROM ASME-ACI 395-74, SECTION III, DIV. 2, 1977, ARTICLE CC-3421.6 "PERIPHERAL SHEAR"



ACI 349 & 359:

$$\frac{v_u}{\sqrt{f'_c}} = 4\sqrt{1 + \frac{f_m}{4\sqrt{f'_c}}}$$

where $f_m = \rho f_s$

WHERE f_m IS TAKEN AS ρf_s

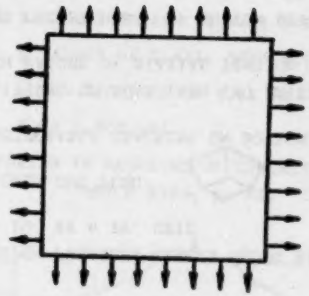
f'_c = 3500 PSI

f_y = 60 KSI

AND v_u IS CALCULATED AT d/2 FROM FACE OF LOADED AREA

(2)

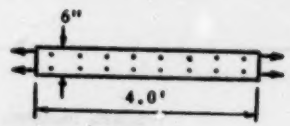
SPECIMEN CONFIGURATION



TENSION APPLIED TO REINFORCING STEEL.

TWO LAYERS IN EACH DIRECTION AT 6 INCH SPACING.

(3)



SLAB GEOMETRY

MATERIALS

-- GRADE 60 REBARS

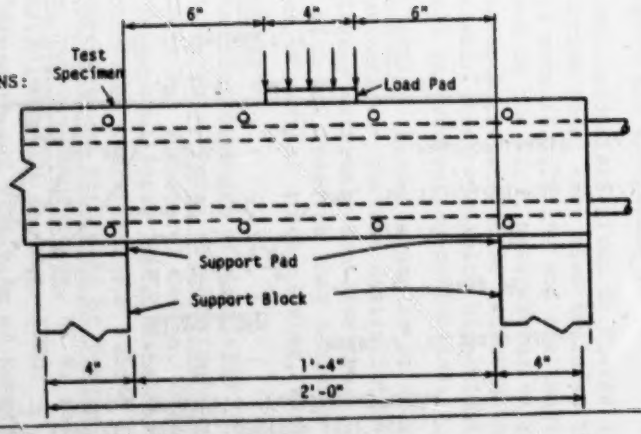
-- ρ = 1.44% IN ONE DIRECTION
ρ = 3.17% IN OTHER DIRECTION

-- f'_c RANGED FROM 3200 PSI TO 4500 PSI

-- CRUSHED LIMESTONE AGGREGATE, 1 IN. MAX.

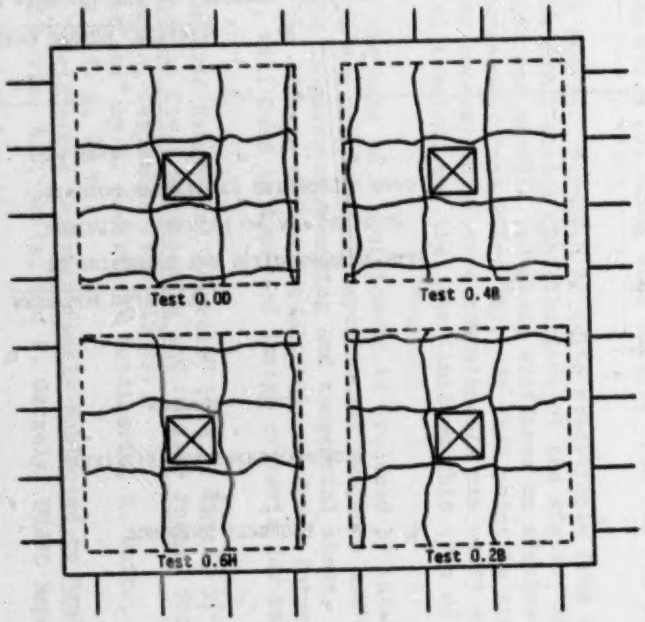
(4)

LOAD AND SUPPORT CONDITIONS:



5

CRACKING PATTERNS AND LOAD LOCATIONS:



6

(Dashed lines indicate centerlines of support, ■ indicates location of punching load)

TEST CONDITIONS AND PARAMETERS

- -- ALL SPECIMENS PRECRACKED IN BIAXIAL TENSION AT 0.6 TO 0.7 f_y
- -- APPLIED BIAXIAL TENSION DURING PUNCHING RANGED FROM 0 TO 0.8 f_y
- -- LOAD THROUGH 4 IN. SQUARE STEEL PAD LOCATED BETWEEN REBARS
- -- STATIC LOADING ONLY
- -- 27 PUNCHING TESTS

7

BEHAVIOR:

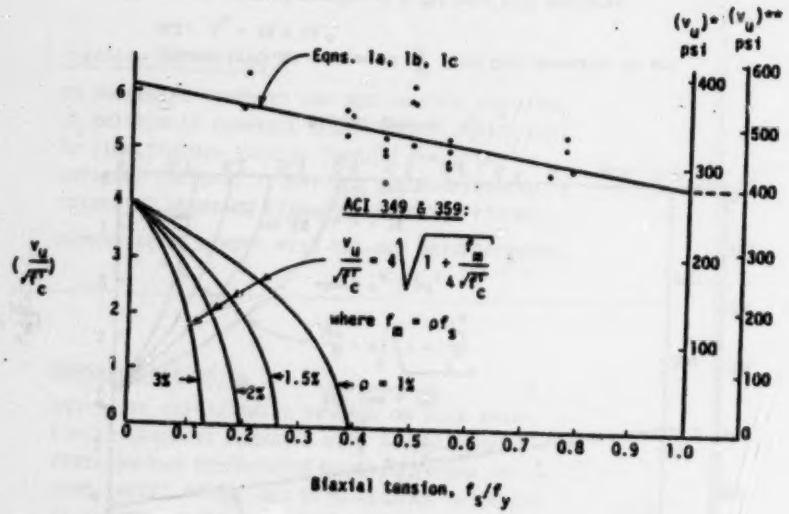
- -- TENSION PRODUCES CRACKS ALONG BARS IN A 6 IN. BY 6 IN. GRID
- -- PUNCHED OUT PLUG:
- -- CONSIDERABLE CRACKING ON BOTTOM FACE
- -- SPLITTING OF SLAB INTO TWO LAYERS FOR HIGH VALUES OF BIAXIAL TENSION
- -- SOME "STRETCHING" OF BOTTOM REBARS

8

$$v_u \text{ (in psi)} = 400 + 0.8\alpha\rho_{min} f_y (1 - f_s/f_y) \qquad \text{(1a)} \; **$$

$$v_u \text{ (in psi)} = 278 + 0.11\rho_{min} f_y (1 - f_s/f_y) \qquad \text{(1b)} \; *$$

$$\frac{v_u}{\sqrt{f'_c}} = 6.1 - 1.6(f_s/f_y) \qquad \text{(1c)} \; *$$



* assumed critical section at $\frac{d}{2}$ from load boundary, as per ACI; b_o = 34.5 in.

** actual critical section at 0.22d from load boundary; b_o = 24 in.

Shear stress at punching failure vs. biaxial tension ratio

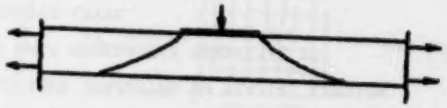
(9)

JOHNSON'S SLAB TESTS:

- -- 3.54 in. THICK SLABS
- -- BIAXIAL TENSION (f_s/f_y) = 0, 0.43, 0.86
- -- ρ VALUES OF 1.42%, BOTH DIRECTIONS
- -- PUNCH DIAMETER OF 4.72 in.
- -- f'_c = 5,500 psi

(REPORTED IN MAGAZINE OF CONCRETE RESEARCH, MARCH 1980, p. 45)

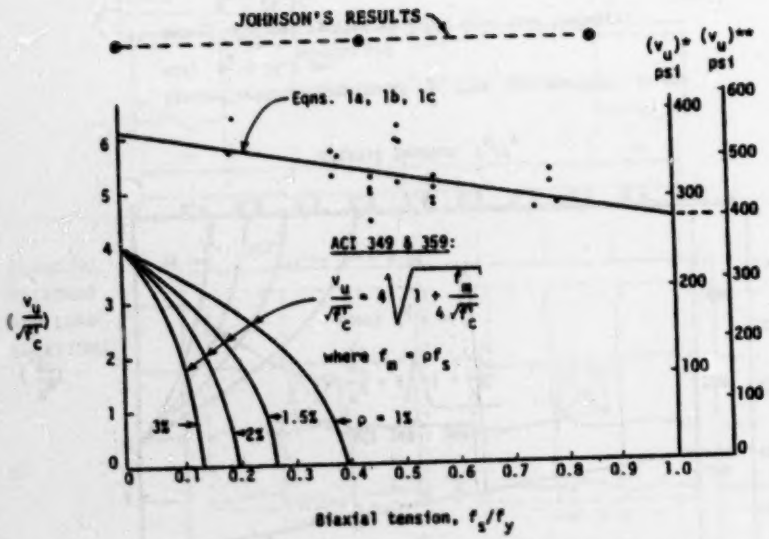
(10)



JOHNSON'S RESULTS:

- -- ALL THREE SPECIMENS FAILED AT P = 36.6 kips
- -- BRITISH STANDARD BS 5400, PT. 4, gives P = 18.7 kips
- -- BRAESTRUP'S PLASTIC THEORY GIVES P = 43.2 kips

(11)



* assumed critical section at $\frac{d}{2}$ from load boundary, as per ACI; b_o = 34.5 in.

** actual critical section at 0.22d from load boundary; b_o = 24 in.

Shear stress at punching failure vs. biaxial tension ratio

(12)

COMPARISON WITH UNIAXIAL RESULTS:

FOR FAILURE DEFINED ON 6 IN. BY 6 IN. SQUARE PLUG SURFACE,

$$v_u = 400 + 0.8\alpha\rho_{min} f_y \left(1 - \frac{f_s}{f_y}\right)$$

(13)

WHERE $\alpha = \frac{1}{5}$ AS COMPARED WITH 1.0 FOR SHEAR STRENGTH IN PRESENCE OF UNIAXIAL TENSION

CONCLUSIONS

1. STATIC PUNCHING (PERIPHERAL) SHEAR STRENGTH OF ORTHOGONALLY REINFORCED CONCRETE SUBJECTED TO BIAXIAL TENSION IS ONLY MODERATELY SENSITIVE TO TENSION LEVEL.
2. COMPARISONS ARE NEEDED WITH RESULTS OF THICK SLABS, AND FROM EXPERIMENTS WITH DIFFERENT PARAMETERS.
3. CURRENT DESIGN RULES (ASME SECTION III, DIV. 2, AND ACI 349 and 359) ARE HIGHLY CONSERVATIVE, LEADING TO EXCESSIVE REINFORCING STEEL AND UNNECESSARY CONGESTION PROBLEMS; HENCE DIFFICULTY IN OBTAINING HIGH QUALITY CONCRETE.

(14)

PRESSURIZED REINFORCED CONCRETE NUCLEAR CONTAINMENT STRUCTURES UNDER SEISMIC LOADING

RESEARCH OBJECTIVES

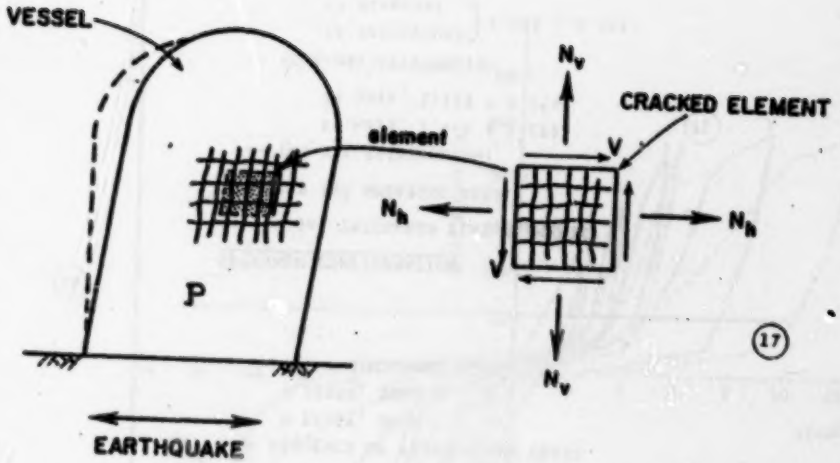
TO DETERMINE STRENGTH AND DEFORMATION BEHAVIOR OF REINFORCED CONCRETE CONTAINMENTS SUBJECTED TO SIMULTANEOUS BIAXIAL TENSION (PRODUCED BY INTERNAL PRESSURIZATION) AND SEISMIC-INDUCED REVERSING MEMBRANE (TANGENTIAL) SHEAR FORCES. (15)

EMPHASIZE SPECIMENS WITH TWO-WAY REINFORCEMENT.

RESEARCH APPROACH

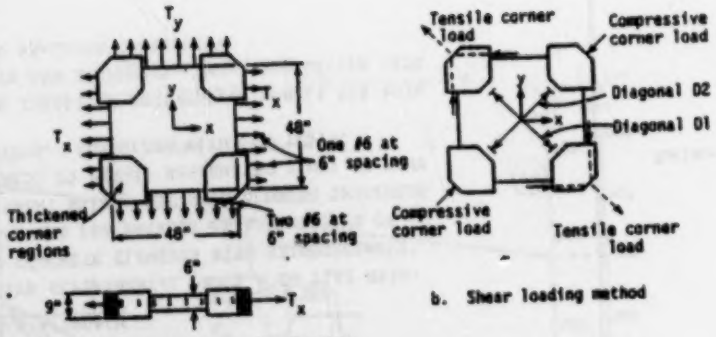
EXTENSIVE EXPERIMENTAL PROGRAM ON FLAT REINFORCED CONCRETE ELEMENTS WITH REINFORCEMENT PATTERNS AND PERCENTAGES REPRESENTATIVE OF CONTAINMENT WALLS, BUT WITH REDUCED THICKNESS (6 INCHES) TO PERMIT SYSTEMATIC STUDY OF MANY PARAMETERS. CORRELATE WITH PCA TESTS. (16)

DEVELOP IDEALIZED ENGINEERING MODELS FOR BOTH STRENGTH AND STIFFNESS, AND EMBED LATTER INTO DYNAMIC ANALYSIS CAPABILITY.



(17)

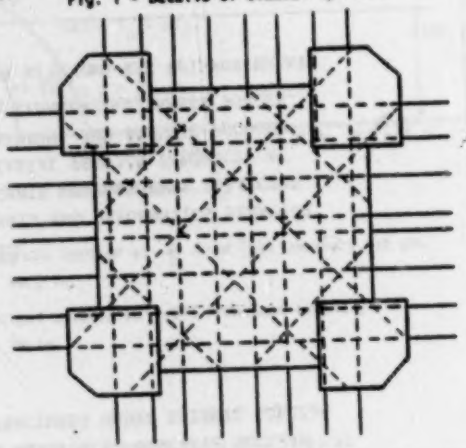
Nuclear containment vessel under combined internal pressurization (p) and seismic forces.



a. Geometry and tensile loading method

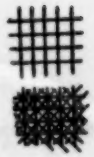
b. Shear loading method

Fig. 1 - Details of biaxial specimen



Specimen incorporating diagonal reinforcing

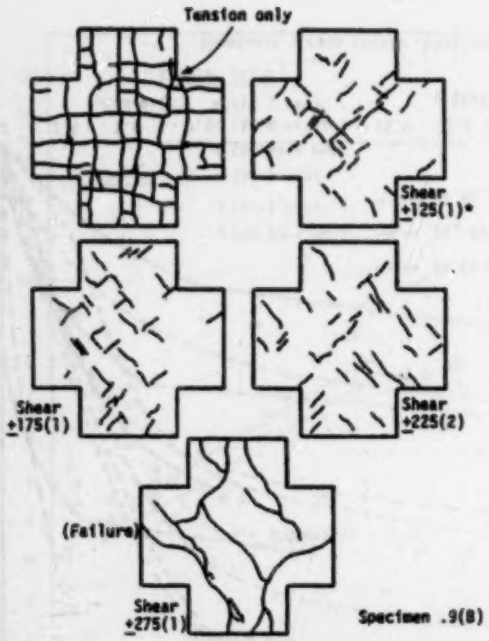
EXPERIMENTAL PARAMETERS



- -- 2- and 4-WAY REINFORCEMENT.
- -- REINFORCEMENT RATIOS.
- -- APPLIED TENSION LEVELS
  0, $0.3f_y$, $0.6f_y$, $0.9f_y$.
- -- SHEAR LOADING: MONOTONIC & TWO TYPES OF CYCLIC.
- -- SHEAR STRESS LEVEL.
- -- SEQUENCE OF TENSION AND SHEAR
  T FIRST, THEN V
  V FIRST, THEN T
  V & T SIMULTANEOUSLY.

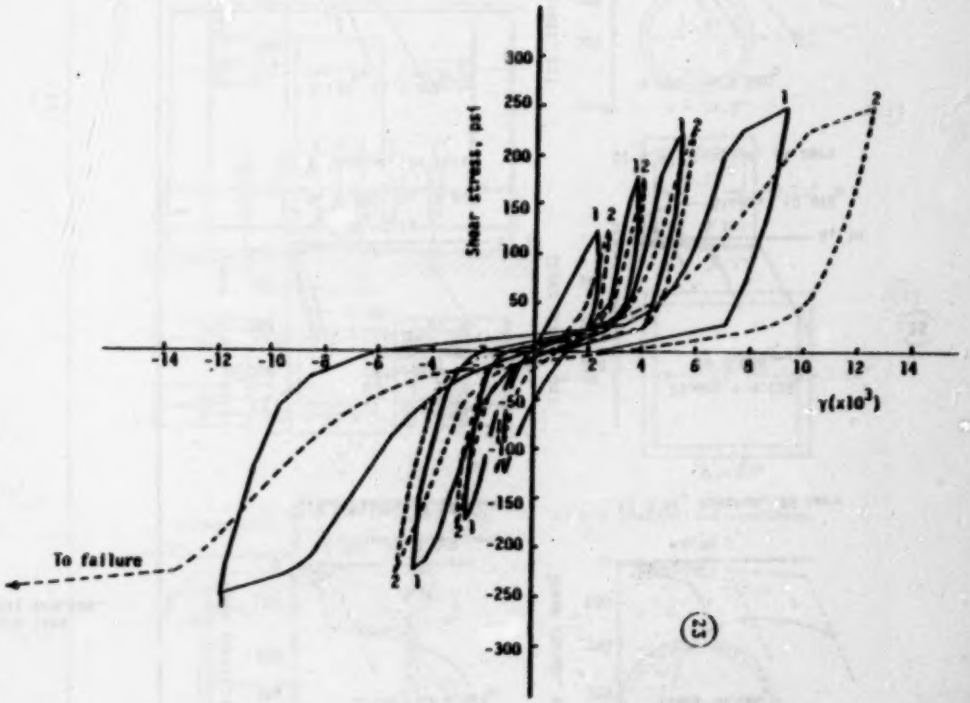
SPECIMEN CONFIGURATION

- -- 6 in. THICKNESS FLAT ELEMENTS.
- -- 288 IN^2 SHEARING AREA.
- -- TWO-WAY REINFORCING:
  #6 bars, 2.44% & 1.22%
  #4 bars, 1.10% & 0.55%.
- -- FOUR-WAY REINFORCING
  #4 orthogonal
  #3 diagonal } 1.30% & 0.76%
- -- GRADE 60 STEEL.
- -- CONCRETE STRENGTH ≈3400 psi.

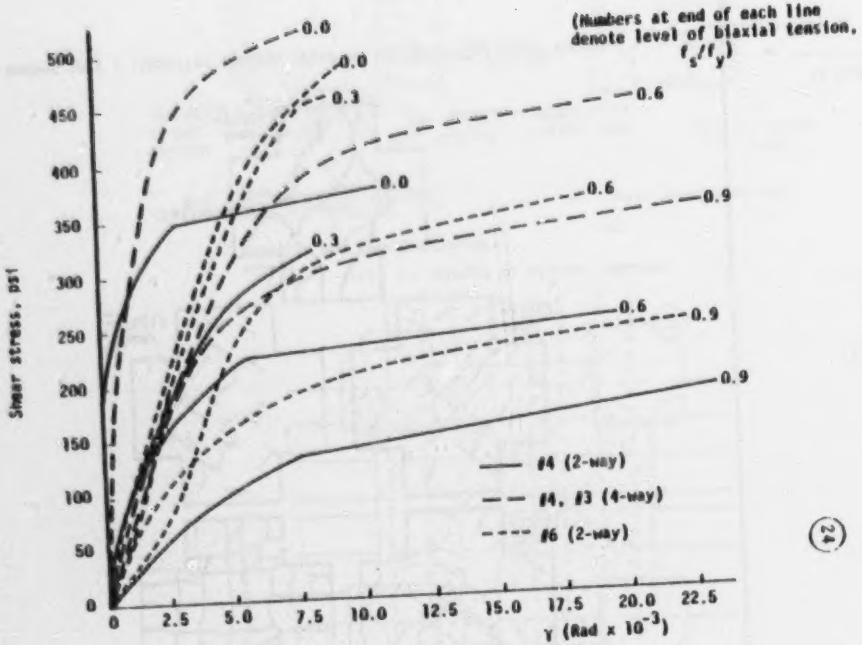


* Number in ( ) signifies loading cycle at the specified shear stress

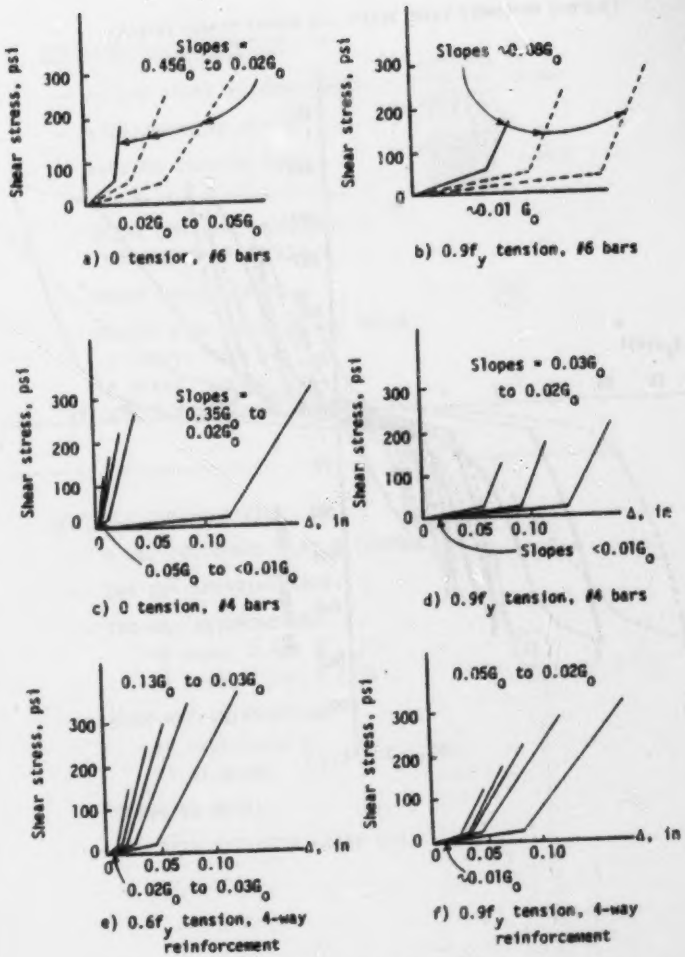
Typical cracking - cyclic test



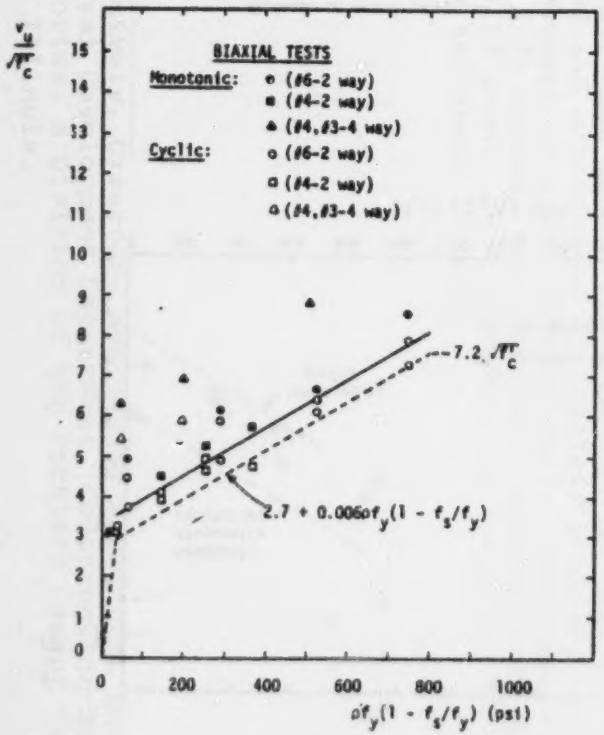
Typical stress-strain for cyclic tests (Specimen 2-0.6-C)



Monotonic stress-strain relationships



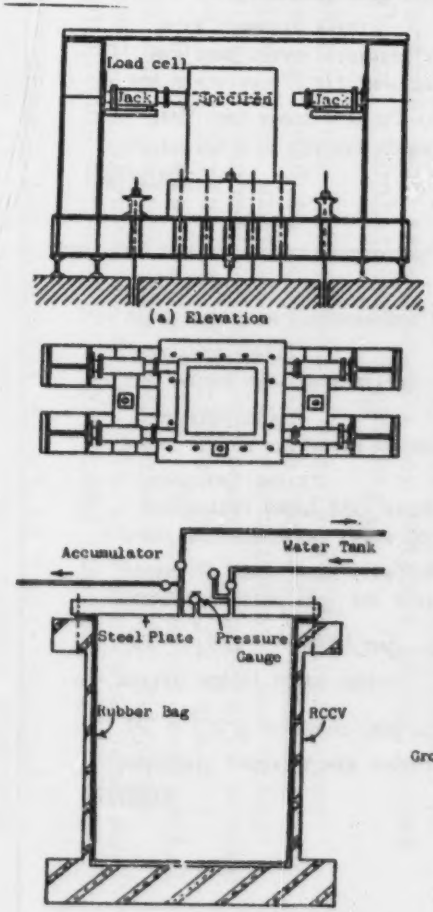
Bilinear stress-strain idealizations

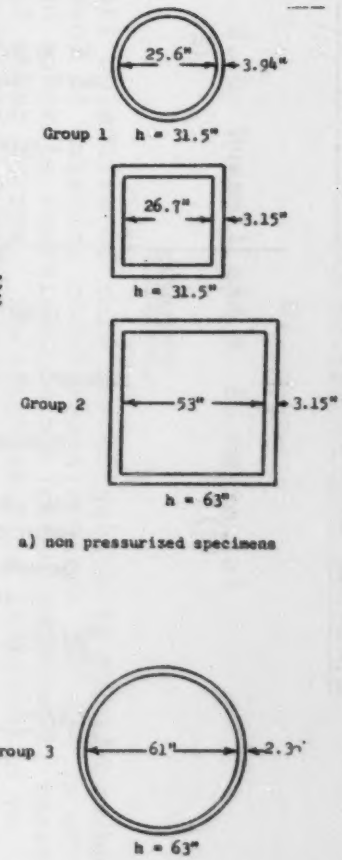


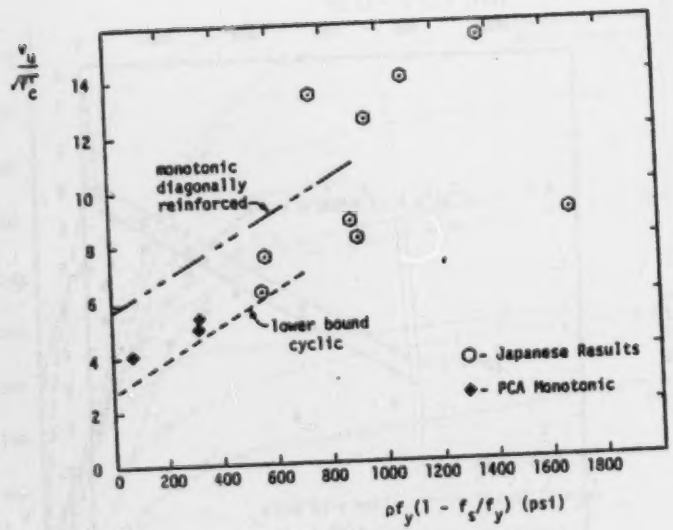
Ultimate strength versus the stress parameter $\rho f_y(1 - f_s/f_y)$; Cornell test results.

JAPANESE TESTS:

- -- IDEALIZED MODELS OF CONTAINMENTS.
- -- 1/10 TO 1/25 SCALE MODELS.
- -- PRESSURIZED AND NON-PRESSURIZED
- -- SHEARING STRENGTH BASED ON FULL SHEARING AREA







Ultimate strength versus $\rho f_y(1 - f_s/f_y)$; PCA and Japanese test results.

(28)

STRENGTH

-- MONOTONIC SHEAR, LOWER BOUND:

$$v_u = [3.3 + 0.006(\rho f_y - \sigma_N)]\sqrt{f'_c}$$

-- CYCLIC SHEAR, LOWER BOUND:

$$v_u = [2.7 + 0.006(\rho f_y - \sigma_N)]\sqrt{f'_c}$$

(CORRESPONDS TO $5\sqrt{f'_c}$ for 1.5% GRADE 60 STEEL WITH $0.6f_y$ TENSION)

-- 4-WAY REINFORCEMENT LEADS TO HIGHER STRENGTHS; ABOUT $2\sqrt{f'_c}$ HIGHER THAN EQUATIONS ABOVE.

-- DOWEL ACTION MECHANISM IS IMPORTANT CONSIDERATION.

-- ALL VALUES ARE CONSERVATIVE IN CONTEXT OF CONTAINMENTS.

-- UPPER LIMIT ON STRENGTH NOT FULLY ESTABLISHED.

(29)

STIFFNESS:

-- EFFECTIVE G OF CRACKED CONCRETE IS VERY LOW; ABOUT 0.05 G_o.

-- FOR G = 0.10 G_o, 1/2 IN. STEEL PLATE HAS SAME SHEAR STIFFNESS AS 50 IN. THICK CONCRETE WALL.

-- SHEAR DISPLACEMENT LIMITATIONS MAY BE NEEDED IN DESIGN PROCESS.

-- SHEAR CYCLING LEADS TO A DECREASE IN EFFECTIVE HYSTERETIC DAMPING.

(30)

LARGE SCALE TESTING OF CONTAINMENT ELEMENTS

by

H. G. Russell and R. G. Oesterle*

SUMMARY

The overall objective of this investigation is to determine the adequacy of existing codes and standards for design of reinforced concrete containments subjected to biaxial tensile forces and shear forces. This information is required to ensure practical and safe designs for seismic shear forces in the wall of a containment structure.

Reinforced concrete containment vessels at nuclear power plants are designed to resist forces caused by internal accident pressure, gravity, severe earthquakes, and other severe natural phenomena. Reinforcement is generally placed in vertical and horizontal directions to resist internal pressure. Other bars are placed at 45° to resist horizontal tangential shear forces from earthquakes. This diagonal reinforcement is expensive, difficult to fabricate, and may cause significant congestion inhibiting concrete placement.

The primary reason for diagonal steel is that internal pressure acting in the containment vessel causes membrane tensile forces that in turn produce horizontal and vertical cracking of the wall. Cracking occurs from pressurization during acceptance tests. The ability of a cracked containment vessel to resist tangential shear forces without diagonal reinforcement has not been verified experimentally.

To investigate behavior of containment vessels, large scale tests have been made. The experimental test set up was designed and constructed to simulate boundary conditions of an element in a containment wall. Specimens were 2-ft thick, 5-ft square, and reinforced orthogonally with No. 18 and No. 14 bars. Each specimen was subjected to biaxial tension and shear loading.

*Respectively, Director, and Senior Structural Engineer, Structural Development Department, Construction Technology Laboratories, a Division of the Portland Cement Association, Skokie, Illinois.

Test variables included level of biaxial tension and monotonic versus reversing load. Objective of the test program was to determine strength and deformation characteristics of a membrane element. Ability of a cracked containment to resist shear through dowel action and aggregate interlock was determined from the large scale tests.

This presentation will describe the experimental program and discuss results obtained to date. Results are being used to evaluate current code requirements.

REPORTS AND PAPERS

1. Oesterle, R.G., and Russell, H.G., "Shear Transfer in Large Scale Reinforced Concrete Containment Elements - Report No. 1." Prepared for U. S. Nuclear Regulatory Commission by Construction Technology Laboratories, NUREG/CR-1374, April 1980, 163 pp.

2. Oesterle, R.G. and Russell, H.G., "Tangential Shear Tests of Reinforced Concrete Containment Elements," Fifth International Conference on Structural Mechanics in Reactor Technology, Berlin, August 1979, Session JK-P, pp 115-156.

3. Oesterle, R.G., and Russell, H.G., "Shear Tests on Reinforced Concrete Membrane Elements," Engineering and Nuclear Power, Vol. I, preprint No. 3594, ASCE, April 1979, 22 pp.

ANALYTICAL APPROACH FOR EVALUATION OF CODES, STANDARDS, AND THE INHERENT SAFETY MARGIN IN SAFETY-RELATED CONCRETE STRUCTURES

by

Oral Buyukozturk
Jerome Connor

Massachusetts Institute of Technology
Cambridge, Mass. 02139

Reinforced concrete is a competitive material for construction of nuclear power plant containment structures. However, a drawback is the complexity of its behavior and the limited behavioral knowledge available to the designer as basis for realistic and safe designs. With the evolution of structural systems for nuclear power plants, there has been some increase in the magnitude of pressure-induced internal forces. More significantly, increasing attention to earthquake design led to higher seismic internal forces which now are a major design consideration. In addition, design loadings have come to include high local loads. These include external loads (e.g., tornado missile impact, turbine fragment) and internal loads associated with pipe whip and fluid jet impact due to postulated pipe rupture. The change in nature, and the increase in magnitudes of forces to be resisted have confronted the designer with conditions that demand extension of his knowledge, based on experiments and accurate analysis procedures.

Accurate analytical procedures for prediction of the stress and deformation fields of homogeneous continua encounter numerous difficulties when applied to reinforced concrete structures. Among these are the incorporation of the nonhomogeneous behavior caused by the presence of the reinforcement, concrete cracking, nonlinear concrete behavior in multiaxial stress states, and shear transfer across cracked concrete sections.

The research at M.I.T. is aimed at developing a numerical procedure for the analysis of complex concrete structures as a basis for safe and rational design. A finite element program ARC for the three dimensional analysis of safety-related nuclear structures has been developed, verified and applied. ARC performs nonlinear incremental analysis. As loads are incrementally applied to the structure stress and strain histories are followed at integration points. With each increment of loads, the constitutive relations are modified to account for nonlinearities resulting from cracking of concrete and from its stress-strain behavior in compressive loadings. In this analysis, concrete is modeled by employing isoparametric solid elements with fifteen symmetrically distributed integration points. Reinforcement is modeled by using two-dimensional eight node thin shell membrane elements. The rebar element provides rigidity in orthogonal directions. The effect of the reinforcing bars are smeared over the spacing of the bars. In this element nine integration points are used.

A major problem is the treatment of cracked sections across which shear stresses are transferred by aggregate interlock (interface shear transfer) and dowel action. In-plane shear stiffness of the containment wall decreases as in-plane shear slips develop locally between the crack faces. Also, development of radial shear slips along the penetrating radial cracks results in the

reduction of containment stiffness in the radial direction which may cause serious problems related to the containment-liner interaction. In this area, early investigations were aimed primarily at providing guidelines on the amount of shear stresses that can be transferred across cracks without excessive deformations (shear slip) along the crack plane.

Recently, a new analytical model has been developed at M.I.T. and implemented in ARC which allows for realistic assessment of shear stiffness for cracked reinforced concrete. This predictive model accounts for the effects of important parameters such as initial crack width, crack spacing, dowel and aggregate interlock on the variation of concrete shear modulus. The model was used in the finite element analysis of containment wall specimens tested at PCA. Through a series of sensitivity studies it was found that in the case where 'orthogonal'cracking is maintained throughout the shear application, the shear deformation and reinforcing bar stresses are highly sensitive to the specified shear modulus. For this case, specifying the cracked concrete shear stiffness $G_{cr}=0.5G_e$ as is typically done produces a very stiff shear stress-shear deformation response. If G_{cr} is to be defined as a percentage of G_e (elastic shear modulus) it should be taken as 10% or less for orthogonally reinforced wall portions exhibiting orthogonal through cracking. For models with diagonal cracking, the shear deformation and change in bar stresses were nearly independent of the specified shear rigidity. In actual test specimens, at shear stress levels approximately below 100 psi slipping across orthogonal cracks (i.e., primary mode of shear transfer by aggregate interlock and dowel action) governs the shear rigidity of the specimen. After diagonal cracking occurs (~100 psi shear stress), there is a transition of behavior to a tension tie-compression strut system.

The numerical model developed at M.I.T. has later been extended to represent the cyclic loading conditions. On the bases of controlling parameters from experiments a degradation law is established for interface shear transfer. This degradation law is applied to tests with single crack plane and panel tests with orthogonal crack planes. However, further development of this model and its verification is necessary for a full scale response analysis of safety-related concrete structures. The review of the reported studies indicate that, at present, the major drawback of the existing nonlinear dynamic analysis capabilities for reinforced concrete is the inability to account for realistic laws of material behavior. Therefore extension of the above described material models to dynamic and impulsive loading conditions is essential. Implementation of such models in a nonlinear dynamic analysis program such as DYNACON developed at M.I.T. would provide a powerful tool for a realistic assessment of safety margin in safety-related concrete structures.

Related References

1. O.Buyukozturk and J.Connor, "ARC, A Computer Program for the Three-Dimensional Finite Element Analysis of Reinforced, Prestressed, and Refractory Concrete Analysis", Research Report R78-33, M.I.T., Dept. of Civil Engrg., October 1978.

2. P.Leombruni, O.Buyukozturk and J.Connor, "Analysis of Shear Transfer in Reinforced Concrete with Application to Containment Wall Specimens", NRC Report, NUREG/CR-1085, October 1979.

3. Preliminary Report on "Analysis of Shear Transfer in Reinforced Concrete Under Monotonic and Cyclic Loadings", M.I.T., Dept. of Civil Engrg., July 1980.

4. O.Buyukozturk, J.Connor and P.Leombruni, "Research on Modeling Shear Transfer in Reinforced Concrete Nuclear Structures", Nuclear Engineering and Design, Vol. 59, August 1979.

5. O.Buyukozturk and J.Connor, "Nonlinear Dynamic Response of Reinforced Concrete Under Impulsive Loading:Research Status and Needs", Nuclear Engineering and Design, Vol. 50, October 1978.

ANALYTICAL APPROACH FOR EVALUATION OF CODES, STANDARDS, AND THE INHERENT SAFETY MARGIN IN SAFETY-RELATED CONCRETE STRUCTURES

by

Oral Buyukozturk
Jerome Connor

Massachusetts Institute of Technology
Cambridge, Mass. 02139

Reinforced concrete is a competitive material for construction of nuclear power plant containment structures. However, a drawback is the complexity of its behavior and the limited behavioral knowledge available to the designer as basis for realistic and safe designs. With the evolution of structural systems for nuclear power plants, there has been some increase in the magnitude of pressure-induced internal forces. More significantly, increasing attention to earthquake design led to higher seismic internal forces which now are a major design consideration. In addition, design loadings have come to include high local loads. These include external loads (e.g., tornado missile impact, turbine fragment) and internal loads associated with pipe whip and fluid jet impact due to postulated pipe rupture. The change in nature, and the increase in magnitudes of forces to be resisted have confronted the designer with conditions that demand extension of his knowledge, based on experiments and accurate analysis procedures.

Accurate analytical procedures for prediction of the stress and deformation fields of homogeneous continua encounter numerous difficulties when applied to reinforced concrete structures. Among these are the incorporation of the nonhomogeneous behavior caused by the presence of the reinforcement, concrete cracking, nonlinear concrete behavior in multiaxial stress states, and shear transfer across cracked concrete sections.

The research at M.I.T. is aimed at developing a numerical procedure for the analysis of complex concrete structures as a basis for safe and rational design. A finite element program ARC for the three dimensional analysis of safety-related nuclear structures has been developed, verified and applied. ARC performs nonlinear incremental analysis. As loads are incrementally applied to the structure stress and strain histories are followed at integration points. With each increment of loads, the constitutive relations are modified to account for nonlinearities resulting from cracking of concrete and from its stress-strain behavior in compressive loadings. In this analysis, concrete is modeled by employing isoparametric solid elements with fifteen symmetrically distributed integration points. Reinforcement is modeled by using two-dimensional eight node thin shell membrane elements. The rebar element provides rigidity in orthogonal directions. The effect of the reinforcing bars are smeared over the spacing of the bars. In this element nine integration points are used.

A major problem is the treatment of cracked sections across which shear stresses are transferred by aggregate interlock (interface shear transfer) and dowel action. In-plane shear stiffness of the containment wall decreases as in-plane shear slips develop locally between the crack faces. Also, development of radial shear slips along the penetrating radial cracks results in the

reduction of containment stiffness in the radial direction which may cause serious problems related to the containment-liner interaction. In this area, early investigations were aimed primarily at providing guidelines on the amount of shear stresses that can be transferred across cracks without excessive deformations (shear slip) along the crack plane.

Recently, a new analytical model has been developed at M.I.T. and implemented in ARC which allows for realistic assessment of shear stiffness for cracked reinforced concrete. This predictive model accounts for the effects of important parameters such as initial crack width, crack spacing, dowel and aggregate interlock on the variation of concrete shear modulus. The model was used in the finite element analysis of containment wall specimens tested at PCA. Through a series of sensitivity studies it was found that in the case where 'orthogonal'cracking is maintained throughout the shear application, the shear deformation and reinforcing bar stresses are highly sensitive to the specified shear modulus. For this case, specifying the cracked concrete shear stiffness $G_{cr}=0.5G_e$ as is typically done produces a very stiff shear stress-shear deformation response. If G_{cr} is to be defined as a percentage of G_e (elastic shear modulus) it should be taken as 10% or less for orthogonally reinforced wall portions exhibiting orthogonal through cracking. For models with diagonal cracking, the shear deformation and change in bar stresses were nearly independent of the specified shear rigidity. In actual test specimens, at shear stress levels approximately below 100 psi slipping across orthogonal cracks (i.e., primary mode of shear transfer by aggregate interlock and dowel action) governs the shear rigidity of the specimen. After diagonal cracking occurs (~100 psi shear stress), there is a transition of behavior to a tension tie-compression strut system.

The numerical model developed at M.I.T. has later been extended to represent the cyclic loading conditions. On the bases of controlling parameters from experiments a degradation law is established for interface shear transfer. This degradation law is applied to tests with single crack plane and panel tests with orthogonal crack planes. However, further development of this model and its verification is necessary for a full scale response analysis of safety-related concrete structures. The review of the reported studies indicate that, at present, the major drawback of the existing nonlinear dynamic analysis capabilities for reinforced concrete is the inability to account for realistic laws of material behavior. Therefore extension of the above described material models to dynamic and impulsive loading conditions is essential. Implementation of such models in a nonlinear dynamic analysis program such as DYNACON developed at M.I.T. would provide a powerful tool for a realistic assessment of safety margin in safety-related concrete structures.

Related References

1. O.Buyukozturk and J.Connor, "ARC, A Computer Program for the Three-Dimensional Finite Element Analysis of Reinforced, Prestressed, and Refractory Concrete Analysis", Research Report R78-33, M.I.T., Dept. of Civil Engrg., October 1978.

2. P.Leombruni, O.Buyukozturk and J.Connor, "Analysis of Shear Transfer in Reinforced Concrete with Application to Containment Wall Specimens", NRC Report, NUREG/CR-1085, October 1979.

3. Preliminary Report on "Analysis of Shear Transfer in Reinforced Concrete Under Monotonic and Cyclic Loadings", M.I.T., Dept. of Civil Engrg., July 1980.

4. O.Buyukozturk, J.Connor and P.Leombruni, "Research on Modeling Shear Transfer in Reinforced Concrete Nuclear Structures", Nuclear Engineering and Design, Vol. 59, August 1979.

5. O.Buyukozturk and J.Connor, "Nonlinear Dynamic Response of Reinforced Concrete Under Impulsive Loading:Research Status and Needs", Nuclear Engineering and Design, Vol. 50, October 1978.

LOAD COMBINATIONS

by

Bruce Ellingwood
Research Structural Engineer
National Bureau of Standards
Washington, DC 20234

Statement of Problem

Standard Review Plans for the design of steel and reinforced concrete containments and Category I nuclear structures contain a series of structural load and resistance specifications to insure acceptable performance in the event of extreme environmental or postulated accident events of the form

$$\text{Design resistance} \geq \text{Effect of design loads}$$

These criteria include safety factors in the form of allowable stresses or load and resistance factors in recognition of the nondeterministic nature of the design variables and the need to account for uncertainties in design. In the past, these safety factors have been set on the basis of subjective engineering judgment. They may not lead to consistent reliability or to performance consistent with the objectives of the regulatory agency because they do not consider uncertainty explicitly or consistently. There is no assurance that all the load combinations are necessary or that the safety factors are appropriate in all instances. Finally, the lack of a methodology for adjusting the criteria as additional data become available makes it difficult to establish the audit trail necessary in developing safety provisions for nuclear structures.

Probability theory and structural reliability provide a formal framework for treating design uncertainty and for selecting safety factors that are consistent with a desired level of performance. A collaborative program between Brookhaven National Laboratory and the National Bureau of Standards was begun in mid-1980 with the objectives of (1) developing a rational approach for reliability based design of containment and Category I nuclear structures and (2) to develop specific probability based load factors and load combinations for design of these structures.

Approach

The basic steps in developing probability based design criteria are (1) develop models of structural system behavior; (2) establish a statistical data base for load and strength variables; (3) develop a reliability analysis procedure, and (4) recommend specific probability based load factors and load combinations for design. The analysis of loads and load combinations is a significant aspect of these interrelated steps.

Structural loads on nuclear structures arise from the environment, and operating conditions, and postulated accidents. In general, these loads are described as random processes. Studies are underway to statistically characterize the loads, including sources of uncertainty in the basic loads and load modeling. The probability distributions of the maximum load during the operating life of the structure, the arbitrary point-in-time load, the mean rate of occurrence of discrete changes in load or of accidental events, and the duration of various loads may all be important in load combination studies. Some of the environmental data used in developing probability based criteria for ordinary buildings can be used here also. Additional analysis of existing data on tornado wind speeds and occurrences is required to obtain information that is useful for reliability calculations. Very little statistical data are available to describe loads arising from operating conditions or postulated accidents.

In load combination analysis, the maximum of a sum of random processes is of particular interest. Since the loads generally vary significantly in time, the maximum of the sum is usually much less than the sum of the maxima of the individual loads. Existing criteria account for this by using "probability" factors on the combined effect or by increasing allowable stresses, and it is here that significant risk inconsistencies may be introduced into load criteria. Approximate solutions for the probability distributions for the maximum of a combination on loads have been developed based on mean upcrossing rates. These approximate solutions appear to be close to the exact solutions for the low exceedance probabilities of interest in nuclear structures.

Practical probability based loading criteria can be developed which are no more complex than existing procedures and yet have a well documented rationale. This may be done through an optimizing process. A safety checking format is first selected considering the conflicting needs for theoretical rigor and simplicity. In conventional structures, the absolute sum of loads has usually been used; however, other formats may be appropriate for nuclear structures in which multiple dynamic forces may be combined. Having selected a format, the safety factors are adjusted to minimize the deviation between the limit state probabilities and the target probability for all likely design situations. This procedure has already been used to develop practical probability based loading criteria for ordinary structures. Its application to nuclear structures, while complicated by the number of loads and their dynamic nature, presents no particular conceptual problem. There are numerous advantages to specifying the load side of the checking equation to be independent of construction material. The possibility of doing this for nuclear structures will be investigated at a later phase in the study.

LOAD COMBINATIONS

OBJECTIVES

DETERMINE APPROPRIATE LOAD COMBINATIONS FOR DESIGN OF NUCLEAR STRUCTURES

DETERMINE SAFETY FACTORS CONSISTENT WITH RELIABILITY OBJECTIVES

SCOPE

STRUCTURAL SYSTEMS COVERED BY S.R.P. 3.8.1 - 3.8.4

SAFETY CHECKING

DESIGN RESISTANCE > EFFECT OF DESIGN LOADS

S.R.P. 3.8.4 - CATEGORY I STRUCTURES

Allowable stress design (steel)

(2a) $1.5F_aS = D + L + T_o + R_o + E$

Strength Design (concrete)

(2b) $\phi R_u = 0.75(1.4D + 1.7L + 1.9E + 1.7R_o + 1.7T_o)$

CURRENT LOAD CRITERIA

DEPEND ON SUBJECTIVE JUDGMENT AND EXPERIENCE

DO NOT CONSIDER RANDOMNESS EXPLICITLY OR CONSISTENTLY

MAY NOT LEAD TO CONSISTENT RELIABILITY AND PERFORMANCE

LOAD CRITERIA - SPECIAL CONSIDERATIONS FOR NUCLEAR STRUCTURES

NUMBER OF LOADS AND LOAD COMBINATIONS IS LARGE

CONSEQUENCES OF FAILURE ARE SEVERE

EXPERIENCE WITH EXISTING LOAD CRITERIA IS LIMITED

COMBINATIONS MAY INVOLVE MULTIPLE DYNAMIC LOADS

PROBABILISTIC DESCRIPTION OF EXTREME LOADS IS UNCERTAIN

PROBABILISTIC LIMIT STATES DESIGN

LIMIT STATE - A CONDITION WHERE A STRUCTURE OR ELEMENT BECOMES UNFIT FOR ITS PURPOSE

- ULTIMATE LIMIT STATES - RUPTURE, INSTABILITY, LOSS OF EQUILIBRIUM
- SERVICEABILITY LIMIT STATES - DEFLECTION, VIBRATION, FATIGUE

RELIABILITY - THE PROBABILITY OF NOT ACHIEVING ANY LIMIT STATE

ADVANTAGES OF PROBABILISTIC LIMIT STATES DESIGN

- INCREASES AWARENESS OF SAFETY AND SERVICEABILITY
- IMPROVES FEEL FOR LOADS AND BEHAVIOR
- SIMPLIFIES DESIGN BY ENCOURAGING SAME PROCEDURE FOR ALL CONSTRUCTION MATERIALS
- PROVIDES TOOL FOR EXERCISING JUDGMENT
- PROVIDES RATIONAL MEANS FOR UPDATING STANDARDS
- ENABLES MORE ECONOMICAL DESIGNS BY REDUCING PRESENT LARGE RANGE IN DESIGN RELIABILITY

INFORMATION FOR RELIABILITY BASED DESIGN

- Reliability analysis procedure
- Statistical characterization of each variable
  - Probability distribution
  - Mean and variance
- Limit states
- Checking equations

STRUCTURAL RELIABILITY

LIMIT STATE

$$g(X_1, X_2, \ldots, X_m) = 0$$

Principles of mechanics

Experimental data

RELIABILITY (LIMIT STATE PROBABILITY)

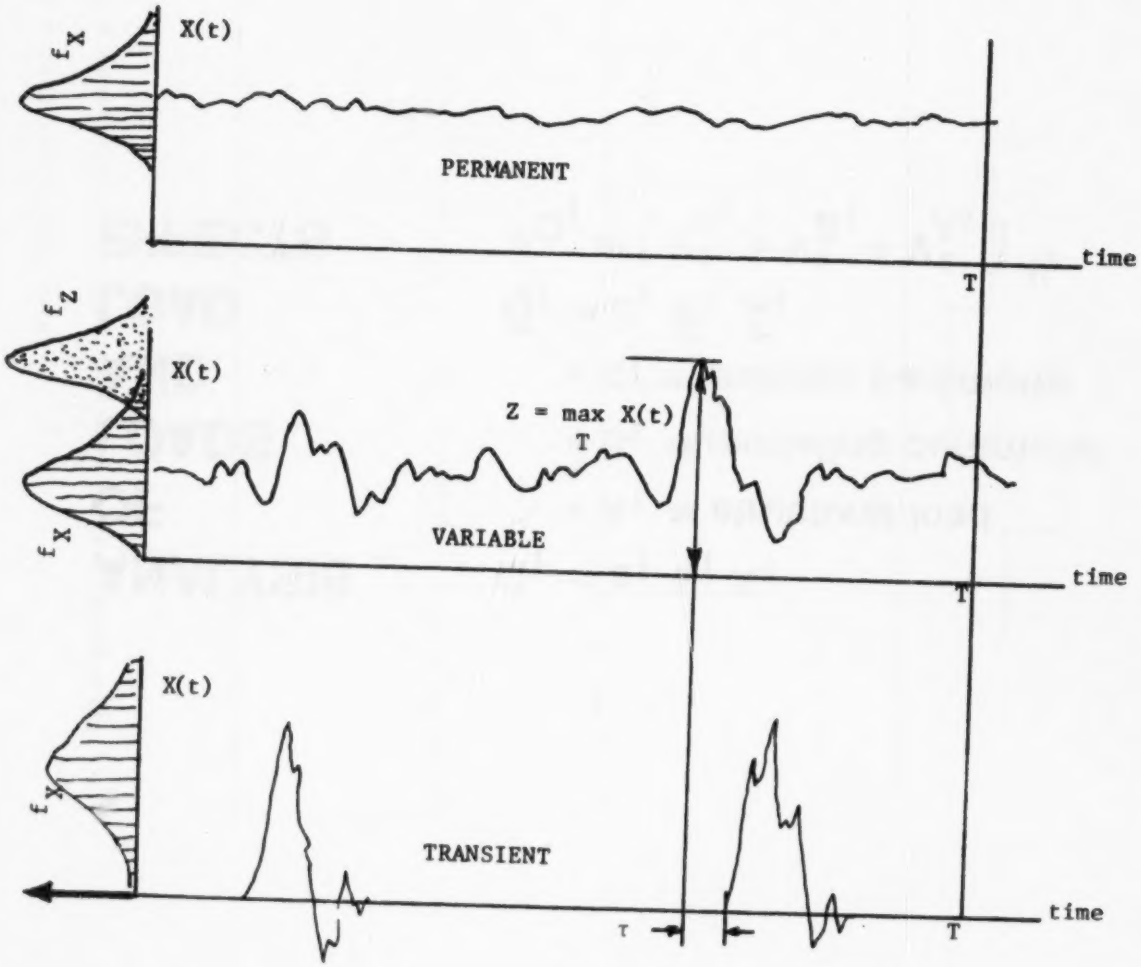
$$p_f = \int_{g<0} \cdots \int f_X(x_1, x_2, \ldots, x_m)\, dx_1 dx_2 \ldots dx_m$$

Limit state

Load Combination

Structural member and material

STRUCTURAL LOADS



ANALYSIS OF LOADS AND LOAD EFFECTS

$$Q_i = c_i \, B_i \, A_i$$

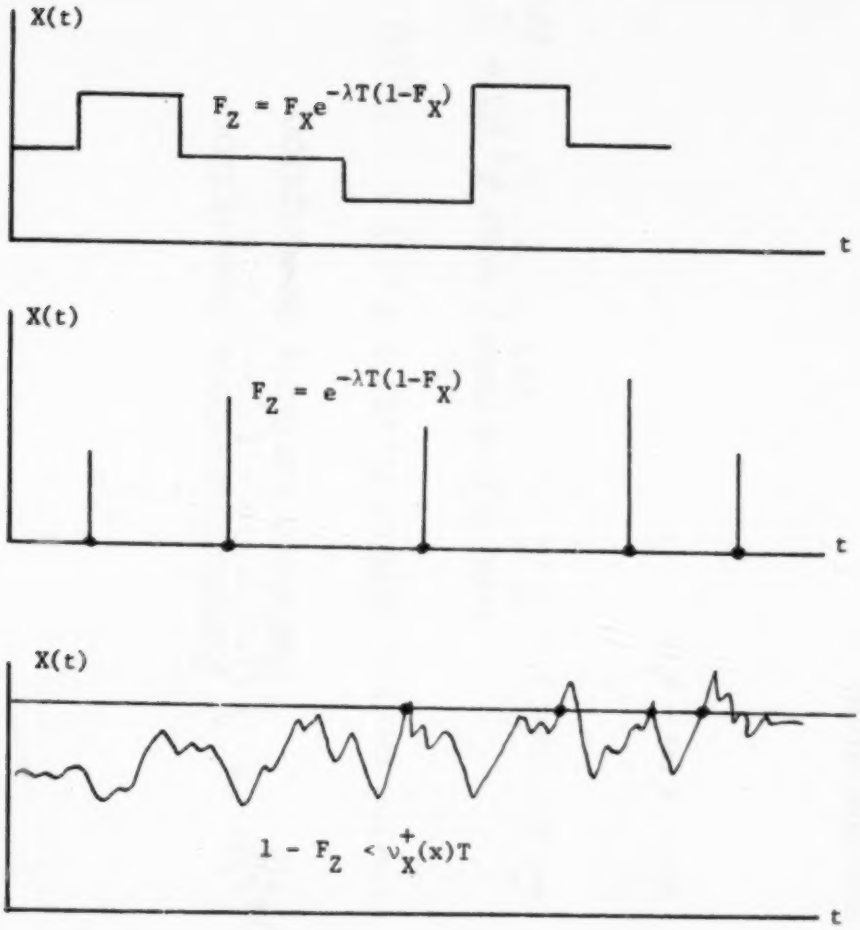
- A_i = structural load
- B_i = modeling parameter
- c_i = analysis parameter

$$\bar{Q}_i = \bar{c}_i \, \bar{B}_i \, \bar{A}_i$$

$$V_{Q_i} = \left[V_{c_i}^2 + V_{B_i}^2 + V_{A_i}^2 \right]^{1/2}$$

STRUCTURAL LOAD MODELS

$$Z = \max_{T} X(t)$$



F_X = arbitrary point-in-time probability distribution

λ = Mean recurrence rate

$\nu_X^+(x)$ = Upcrossing rate of level x

LOAD COMBINATIONS

$$X(t) = X_1(t) + X_2(t)$$

$$\max_T X(t) \leq \max_T X_1(t) + \max_T X_2(t)$$

$$\text{Prob}[\max X(t) > x] \leq \nu_X^+(x)T$$

$$\nu_X^+(x) \leq \int \nu_1^+(u) f_2(x-u)\,du + \int \nu_2^+(x-u) f_1(u)\,du$$

LOAD COMBINATIONS

- Random process description
- Random variable description

$$Z(t) = X_1(t) + X_2(t) + \dots X_n(t)$$

$$\max_T Z(t) \approx \max_{i=1}^{n} \left[\max_T X_i(t) + \sum_{j \neq i} X_j(t) \right]$$

FORMAT FOR CHECKING EQUATION

- Theoretical appeal
- Simplicity
- Similarity with existing procedures

LOAD FACTOR FORMATS

- $U = \gamma\,(D + L + W + \ldots)$
- $U = \gamma_D D + \psi\,(\gamma_L L + \gamma_W W + \gamma_T T)$
- $U = \gamma_D D + \gamma_Q \left[Q_i + \sum_{\substack{j=1 \\ j \neq i}}^{n} \psi_{o_j} Q_j\right]$
- $U = \gamma_D D + \gamma_Q\, Q_{max} + \gamma_{apt}\, Q_{apt}$

RESISTANCE FACTORS

Structural action

$R = \phi R_n$

- Member strength
- Limit state
  - Mode of failure
  - Consequence of failure
- Material design specification

Material strengths, member dimensions

$f_d = f_n / \gamma_m$

$a_d = a_n \pm \Delta$

- Same for all limit states
- Individual sources of variability considered

DESIGN CRITERIA

SELECT FORMAT

$$\phi R = \sum \gamma Q$$

DETERMINE FACTORS ϕ AND γ TO MINIMIZE

$$\sum w_k h[p_f(R_k), p_{f_o}]$$

p_{f_o} = target limit state probability

$p_f(R_k)$ = limit state probability for design R_k

w_k = relative frequency of situation k

h(a,b) = measure of closeness between a and b

ULTIMATE LIMIT STATES

- Factored resistance > Effects of factored loads
- Load and resistance factor design (LRFD)

$$\phi R_n > \sum_i \gamma_i Q_{n_i}$$

ϕ - resistance factor

γ_i - load factor

A58 LOAD FACTORS AND LOAD COMBINATIONS

$$\phi R_n \geq \text{effect of} \begin{cases} 1.4\,(D + T) \\ 1.2\,D + 1.6\,L \\ 1.2\,D + 1.6\,S + 0.5\,L \\ 1.2\,D + 1.3\,W + 0.5\,L \\ 1.2\,D + 1.5\,E + (0.5\,L \text{ or } 0.2\,S) \\ 0.9\,D - (1.3\,W \text{ or } 1.5\,E) \end{cases}$$

D = permanent load
L = live load
S = snow load
W = wind load
E = earthquake load
T = self straining force

LAWRENCE LIVERMORE NATIONAL LABORATORY

LOAD COMBINATION RESEARCH PROGRAM
PROGRAM OVERVIEW

PRESENTED BY

C. K. CHOU

PRESENTATION TO

8TH WATER REACTOR SAFETY RESEARCH INFORMATION METTING

GAITHERSBURG, MARYLAND

OCTOBER 31, 1980

Load Combination Research Program at the Lawrence Livermore National Laboratory*

by

C. K. Chou

Lawrence Livermore National Laboratory
University of California, Livermore, California 94550

SUMMARY

This paper presents an overview of the Load Combination Research Program at the Lawrence Livermore National Laboratory sponsored by the U.S. Nuclear Regulatory Commission (USNRC).

General design criterion 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A, 10CFR50, requires that structures, systems, and components important to the safety of nuclear power plants be designed to withstand combinations of effects of natural phenomena and the effects of normal and accident conditions.

Historically, the USNRC has required that the structural/mechanical responses due to various postulated accident loads and loads caused by natural phenomena be combined in the analysis of structures, systems, and components important to safety. This requirement appears in the various regulatory documents such as regulatory guides, regulation revisions, and branch technical positions, as well as in the standard review plan..

The postulated accident loads and loads caused by natural phenomena such as earthquakes are random events. Amplitude, duration, frequency content, time of occurrence, and time-phase relationship are random and stochastic in nature. Without a systematic probability assessment, it is almost impossible to come up with a rationale for an appropriate combination of events. Requirements to consider dynamic events acting concurrently have therefore been based on judgment tending towards conservatism, because no comprehensive probabilistic study and unified philosophy exists on which to base better-founded decisions. Consequently, safety margins between various systems and from plant to plant vary widely. Trying to follow the inconsistent and fragmented load combination requirements is frustrating to the nuclear industry.

*Presented at the 8th Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, October 31, 1980.

A unified approach to load combination is needed by the NRC as well as the nuclear industry. This approach must evolve from a rational procedure to address the following basic issues:

1. How to postulate pipe break locations.
2. What loads need to be combined?
3. If certain loads must be combined, what is the appropriate methodology to combine them?
4. What factors should be applied to load effects or responses?
5. What are appropriate service levels, load categories, and stress limits?

The Load Combination Program, a multi-million dollar, long-term research program undertaken by the Lawrence Livermore National Laboratory represents a milestone effort funded by the USNRC to resolve these difficult and important issues. Current planning is that the program will continue through 1985. Expenditures for the fiscal year of 1980 are approximately $1.3 million dollars. The first phase effort scheduled to be completed at the end of Summer 1980 has two major objectives to be accomplished: (1) to determine the probability of a large LOCA (Loss of Coolant Accident) induced directly and indirectly by a range of earthquakes; (2) to develop a general methodology for combining generic dynamic loads for nuclear power plant design.

The objective of determining the probability of a large LOCA induced directly by earthquakes requires the development of a deterministic fracture mechanics model for fatigue crack growth. This model incorporated with stochastic inputs of initial crack size distribution, material properties, stress histories, and leak detection probability is used to calculate the probability of pipe leak and break when subjected to the various loading conditions including seismic. The best estimate model and approach were used in the computation. The probability results together with the uncertainties will be used as a basis to determine whether the low probability events such as a large LOCA and SSE should be combined when nuclear structures and components are designed. This model can also be used to evaluate the reliability of piping systems in general. The results of the evaluation will be used to provide technical basis for the USNRC to modify Regulatory Guide 1.46 (Protection Against Pipe Whip Inside Containment.

The objective of evaluating the probability of a large LOCA induced indirectly by earthquakes requires the identification of accident scenarios induced by earthquakes, such as mechanical and electric failures, explosions, missiles, and fires which, in turn, can cause a LOCA. For each scenario the probability is estimated.

The objective of developing a general methodology for combining generic dynamic loads involves the development of component reliability methodology to derive the appropriate load combination requirements for the nuclear power plant design. This methodology, by considering the event probability, nature of the events, structural properties, response characteristics, and other random phenomena, will achieve consistency in component design. This methodology and the developed model can also be used to evaluate the reliability of the existing structure and component design.

Due to the potential impact on licensing decisions, the NRC and the ACRS have requested that a National Peer Review Panel be formed to review our results and conclusions as well as technical detail. The Review Panel is comprised of five experts in the areas of metallurgy and materials, risk and reliability, structural and mechanical analysis, mathematical analysis, and nuclear system. Two review meetings have been scheduled prior to the ACRS meeting in November 1980.

This presentation describes the overview of the Load Combination Program. The consecutive presentations will address the technical approach, evaluation results and applications in more detail.

OVERVIEW OF PRECENTATION

- o INTRODUCTION
- o PROGRAM OVERVIEW
- o PROJECT I - LOCA-SSE DECOUPLING
- o PROJECT II - LOAD COMBINATION METHODOLOGY DEVELOPMENT
- o TECHNICAL PRESENTATION AGENDA

LOAD COMBINATION PROBLEM OVERVIEW

- o HOW TO DEFINE EACH INDIVIDUAL LOAD?
- o HOW TO DETERMINE THE LOAD COMBINATION EQUATIONS FOR NUCLEAR POWER PLANT DESIGN?
  - HOW TO POSTULATE PIPE BREAK LOCATIONS?
  - WHAT LOADS NEED TO BE COMBINED?
  - IF CERTAIN LOADS MUST BE COMBINED, WHAT IS THE MOST APPROPRIATE METHOD TO COMBINE THEM?
  - HOW MANY LOAD COMBINATION EQUATIONS ARE NEEDED?
- o HOW TO DETERMINE THE APPROPRIATE SERVICE LEVELS, LOAD CATEGORIES, AND DESIGN LIMITS?

ONLY TWO MAJOR ISSUES WERE TO BE ASSESSED IN THE FIRST PHASE OF THE PROGRAM
(COMPLETED AT THE END OF AUGUST, 1980)

o EVENT DECOUPLING

TO CALCULATE THE PROBABILITY OF A LARGE LOCA INDUCED BY A RANGE OF EARTHQUAKES FOR A SELECTED PWR PLANT.

- DETERMINE WHETHER OR NOT LOW PROBABILITY EVENTS SUCH AS THE LOADS POSTULATED FOR THE DOUBLE-ENDED GULLOTINE BREAK IN THE REACTOR COOLANT LOOP NEED TO BE COMBINED WITH SSE SEISMIC.

- DEVELOP PROBABILISTIC FRACTURE MECHANICS MODEL TO AID IN POSTULATING PIPE BREAK LOCATIONS.

o LOAD COMBINATION METHODOLOGY DEVELOPMENT

TO DEVELOP A METHODOLOGY, BASED ON COMPONENT AND SYSTEM RELIABILITY, WHICH CAN BE USED TO:

- EVALUATE THE RELIABILITY OF THE EXISTING DESIGN.

- MORE RATIONALLY DETERMINE

  - WHICH LOADS NEED TO BE COMBINED
  - HOW TO COMBINE THEM
  - APPROPRIATE SERVICE LEVELS AND DESIGN ALLOWABLES

LOAD COMBINATION PROGRAM

PHASE I WORK - SCHEDULE SUMMARY

- PHASE I WORK AUTHORIZED MARCH, 1979.
- PHASE I WORK PLAN DEVELOPED - JULY, 1979.
- TECHNICAL WORK STARTED - JULY, 1979.
- PHASE I TECHNICAL WORK COMPLETION - AUGUST, 1980.
- PHASE I FINAL REPORT (DRAFT) - NOVEMBER, 1980.

LOAD COMBINATION PROGRAM

PHASE I WORK - BUDGET SUMMARY

- TOTAL PHASE I BUDGET - $1256K (MARCH, 1979 THRU SEPTEMBER, 1980).
  - $1056K FROM MECHANICAL ENGINEERING RESEARCH BRANCH.
  - $200K FROM METALLURGY AND MATERIALS RESEARCH BRANCH.
- DISTRIBUTION OF FUNDS
  - PROJECT I - EVENT DECOUPLING - 70%
  - PROJECT II - LOAD COMBINATION METHODOLOGY DEVELOPMENT - 30%

LOAD COMBINATION PROGRAM
PHASE I WORK REVIEW SCHEDULE

- o FIRST PEER REVIEW PANEL MEETING
  SEPTEMBER 18 & 19, 1980
- o NRC/DIVISION OF SYSTEM & RELIABILITY RESEARCH REVIEW (RELIABILITY OF PIPING SYSTEMS)
  SEPTEMBER 23, 1980
- o NRC/NRR REVIEW
  SEPTEMBER 30, 1980
- o SENIOR CONSULTANT REVIEW (EVENT DECOUPLING)
  OCTOBER 13, 1980
- o SECOND PEER REVIEW PANEL MEETING
  NOVEMBER 6 & 7, 1980
  LIVERMORE, CALIFORNIA
- o ACRS MEETING - SUBCOMMITTEE ON COMBINATION OF DYNMAIC LOADS.
  NOVEMBER 18 & 19, 1980
  JUPITER, FLORIDA

PROJECT I ACTIVITY

TO CALCULATE THE PROBABILITY OF LEAK OR DOUBLE-ENDED GUILLOTINE BREAK INDUCED DIRECTLY BY A RANGE OF EARTHQUAKES FOR THE PRIMARY LOOP PIPING (HOT LEG, COLD LEG AND CORSS OVER) FOR A SELECTED PWR PLANT - ZION NUCLEAR POWER PLANT UNIT #1.

ZION UNIT #1 IS A PWR-1 WESTINGHOUSE 4-LOOP PLANT LOCATED 40 MILES NORTH OF CHICAGO ON LAKE MICHIGAN SHORELINE.

OWNER	-	COMMONWEALTH EDISON
A/E	-	SARGENT AND LUNDY
NSSS VENDER	-	WESTINGHOUSE
COMMERCIAL OPERATION	-	JUNE 1973
NET MWe	-	1100

WHY NRC WOULD LIKE TO CONSIDER DECOUPLING A LARGE LOCA AND SSE LOADS

- o THE ORIGINAL COMBINATION REQUIREMENT HAS NO RATIONAL BASIS.
- o THE COST OF PLANTS TO DESIGN FOR COMBINED LARGE LOCA AND SSE IS GREAT.
- o SOME OLD PLANTS WERE SIMPLY NOT DESIGNED FOR LOCA & SSE. MODIFICATION IS EITHER EXPENSIVE OR IMPOSSIBLE.
- o DESIGN FOR BOTH LOCA & SSE MAY NOT BE BENEFICIAL BECAUSE IT MAY REDUCE SYSTEM RELIABILITY DURING NORMAL OPERATION.
  - A LARGE LOCA IN THIS EVALUATION IS DEFINED AS A DOUBLE-ENDED GUILLOTINE BREAK IN THE PRIMARY LOOP PIPING SYSTEM, I.E., HOT LEG, COLD LEG, AND CROSS OVER.
  - EVEN IF THE DECOUPLING DECISION IS APPLIED ONLY TO PRIMARY LOOP PIPING FRACTURE INSIDE THE REACTOR CAVITY WALL OF PWR PLANTS IT WILL BE OF GREAT BENEFIT TO THE NUCLEAR INDUSTRY.

TECHNICAL PRESENTATION WILL BE GIVEN IN DETAIL

- o FRACTURE MECHANICS MODEL
- o INPUT VARIABLES
- o FAILURE CRITERION
- o PROBABILISTIC COMPUTATION
- o ADVANCES IN PIPING RELIABILITY MODEL
- o POSSIBLE APPLICATION OF THE DEVELOPED PIPING RELIABILITY MODEL
- o RESULTS AND CONCLUSIONS

PROJECT II ACTIVITY

TO DEVELOP A METHODOLOGY BASED ON COMPONENT AND SYSTEM RELIABILITY WHICH CAN BE USED TO:

- o EVALUATE THE RELIABILITY OF THE EXISTING DESIGN
- o MORE RATIONALLY DETERMINE
  - WHICH LOADS NEED TO BE COMBINED
  - HOW TO COMBINE THEM
  - APPROPRIATE SERVICE LEVELS AND DESIGN ALLOWABLES.

EXAMPLES OF LOAD COMBINATION REQUIREMENT

STEEL STRUCTURE DESIGN (SRP 3.8.4)

$$0.9Y \geq D+L+T_A+R_A+1.25P_A+1.0(Y_J+Y_R+Y_M)+1.25E$$

SPECIFIED BY NRC

MECHANICAL PIPING DESIGN

$$2.25\ S_H \geq OBE + CO + CH + P + D$$

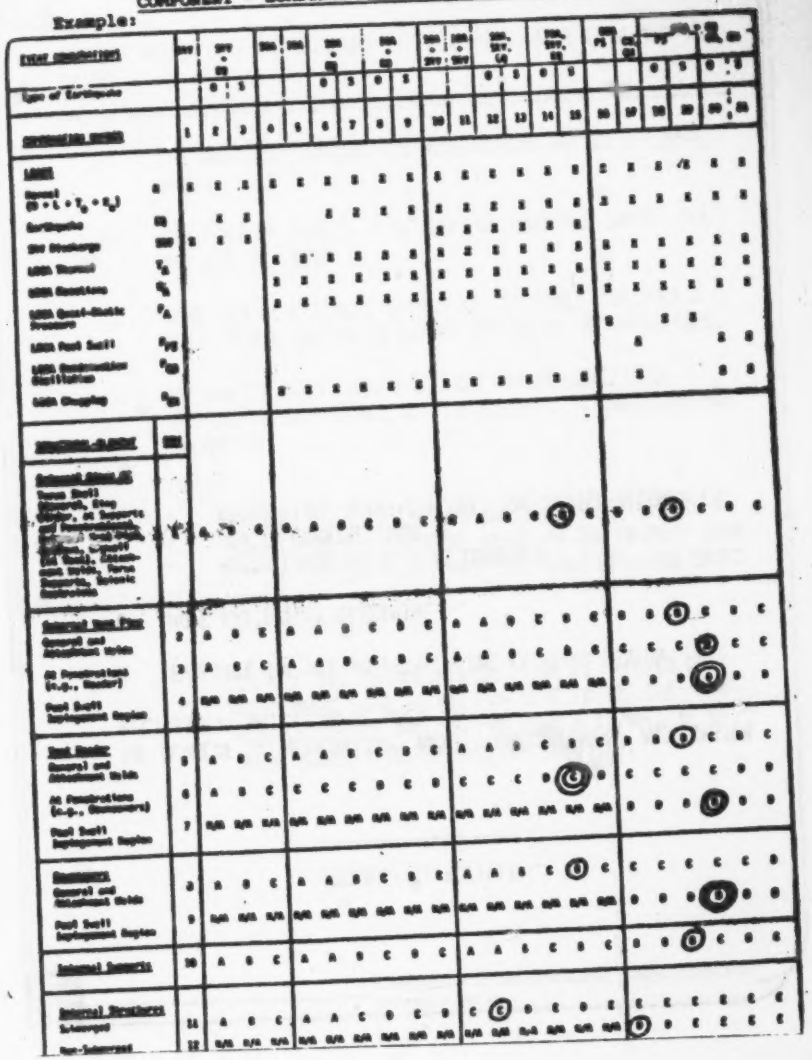
SPECIFIED BY ASME CODE

SPECIFIED BY OWNERS AND THEIR A/E'S APPROVED BY NRC

MARK I CONTAINMENT

COMPONENT - LOADINGS - SERVICE LEVEL ASSIGNMENTS

Example:



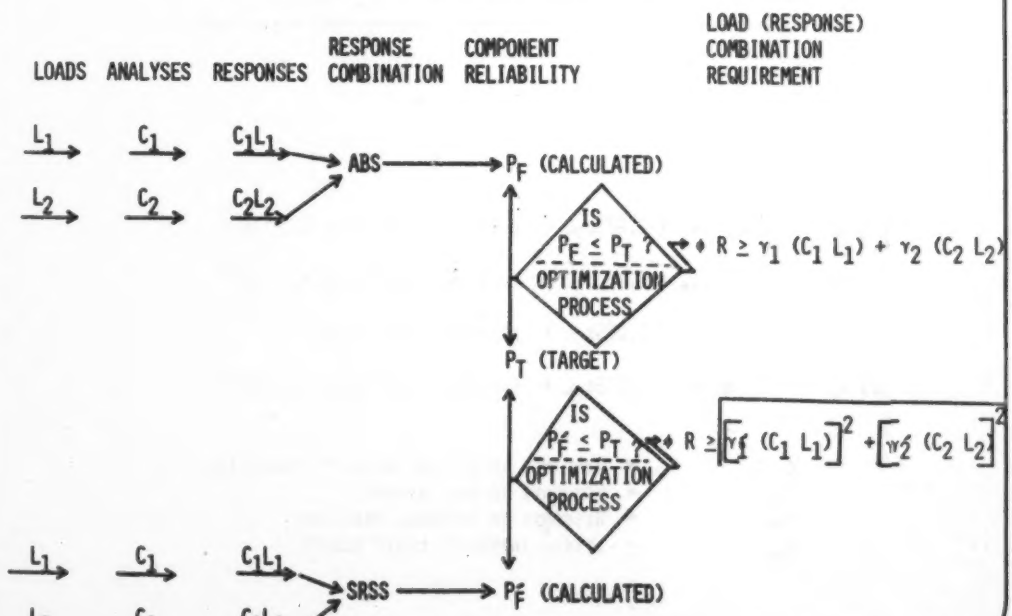
CONCEPT OF USING COMPONENT RELIABILITY METHODOLOGY TO EVALUATE EXISTING DESIGN

LOADS	ANALYSES	RESPONSES	RESPONSE COMBINATION	COMPONENT RELIABILITY
L_1	C_1	C_1L_1	$R \geq C_1L_1 + C_2L_2$	P_F (ABS)
L_2	C_2	C_2L_2		
			COMPARE WITH CRITERIA FOR ACCEPTANCE	
L_1	C_1	C_1L_1	$R \geq \sqrt{(C_1L_1)^2 + (C_2L_2)^2}$	P_F (SSRS)
L_2	C_2	C_2L_2		

LAWRENCE LIVERMORE LABORATORY
University of California

17

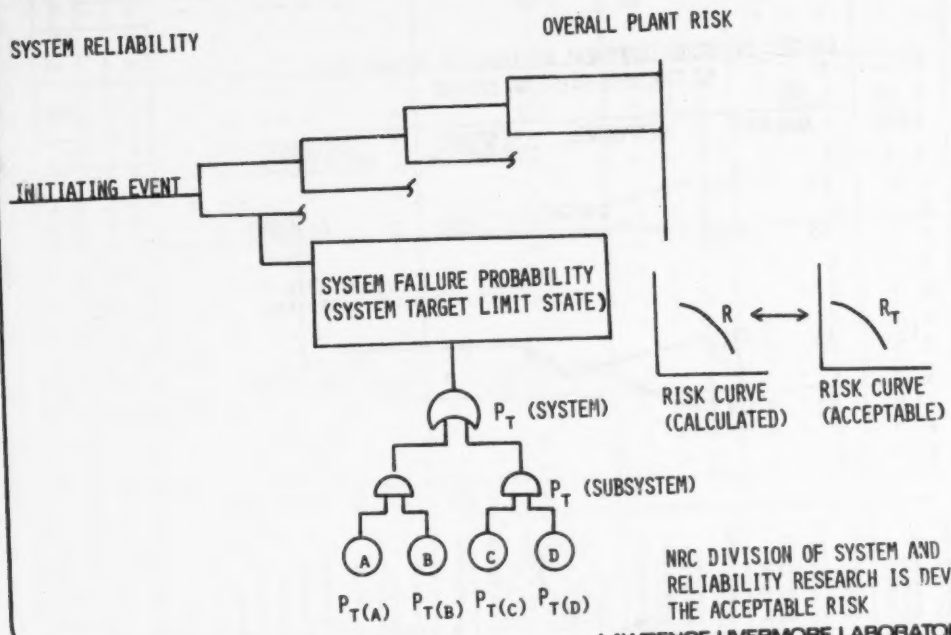
CONCEPT OF USING COMPONENT RELIABILITY METHODOLOGY TO DEVELOP LOAD (RESPONSE) COMBINATION REQUIREMENTS



LAWRENCE LIVERMORE LABORATORY
University of California

18

CONCEPT OF USING SYSTEMS RELIABILITY METHODOLOGY TO DEVELOP COMPONENT TARGET LIMIT STATE



NRC DIVISION OF SYSTEM AND RELIABILITY RESEARCH IS DEVELOPING THE ACCEPTABLE RISK

LAWRENCE LIVERMORE LABORATORY
University of California

THE DESIGNER IS PROVIDED WITH A SERIES OF DETERMINISTIC FACTORED DESIGN EQUATIONS WHICH WILL ENSURE THE ACHIEVEMENT OF A TARGET LIMIT STATE PROBABILITY

EXAMPLE

$$1.5\ s_M = 0.875c_1v_0 + 0.873c_2L + 1.034c_3P \quad \text{FOR T.L.S.P.} = 10^{-2}$$

$$1.5\ s_M = 0.957c_1v_0 + 0.951c_2L + 1.197c_3P \quad \text{FOR T.L.S.P.} = 10^{-4}$$

$$1.5\ s_M = 1.033c_1v_0 + 1.023c_2L + 1.324c_3P \quad \text{FOR T.L.S.P.} = 10^{-6}$$

$$1.5\ s_M = 1.108c_1v_0 + 1.076c_2L + 1.434c_3P \quad \text{FOR T.L.S.P.} = 10^{-8}$$

c_1v_0 = RESPONSE TO INITIAL VELOCITY TRANSIENT
c_2L = RESPONSE TO DEAD WEIGHT
c_3P = RESPONSE TO INTERNAL PRESSURE
s_M = STRESS INTENSITY LIMIT (CODE)

LAWRENCE LIVERMORE LABORATORY
University of California

LAWRENCE LIVERMORE NATIONAL LABORATORY

LOAD COMBINATION RESEARCH PROGRAM
PROGRAM OVERVIEW

PRESENTED BY

C. K. CHOU

PRESENTATION TO

8TH WATER REACTOR SAFETY RESEARCH INFORMATION METTING

GAITHERSBURG, MARYLAND

OCTOBER 31, 1980

Load Combination Research Program at the Lawrence Livermore National Laboratory*

by

C. K. Chou

Lawrence Livermore National Laboratory
University of California, Livermore, California 94550

SUMMARY

This paper presents an overview of the Load Combination Research Program at the Lawrence Livermore National Laboratory sponsored by the U.S. Nuclear Regulatory Commission (USNRC).

General design criterion 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A, 10CFR50, requires that structures, systems, and components important to the safety of nuclear power plants be designed to withstand combinations of effects of natural phenomena and the effects of normal and accident conditions.

Historically, the USNRC has required that the structural/mechanical responses due to various postulated accident loads and loads caused by natural phenomena be combined in the analysis of structures, systems, and components important to safety. This requirement appears in the various regulatory documents such as regulatory guides, regulation revisions, and branch technical positions, as well as in the standard review plan.

The postulated accident loads and loads caused by natural phenomena such as earthquakes are random events. Amplitude, duration, frequency content, time of occurrence, and time-phase relationship are random and stochastic in nature. Without a systematic probability assessment, it is almost impossible to come up with a rationale for an appropriate combination of events. Requirements to consider dynamic events acting concurrently have therefore been based on judgment tending towards conservatism, because no comprehensive probabilistic study and unified philosophy exists on which to base better-founded decisions. Consequently, safety margins between various systems and from plant to plant vary widely. Trying to follow the inconsistent and fragmented load combination requirements is frustrating to the nuclear industry.

*Presented at the 8th Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, October 31, 1980.

A unified approach to load combination is needed by the NRC as well as the nuclear industry. This approach must evolve from a rational procedure to address the following basic issues:

1. How to postulate pipe break locations.
2. What loads need to be combined?
3. If certain loads must be combined, what is the appropriate methodology to combine them?
4. What factors should be applied to load effects or responses?
5. What are appropriate service levels, load categories, and stress limits?

The Load Combination Program, a multi-million dollar, long-term research program undertaken by the Lawrence Livermore National Laboratory represents a milestone effort funded by the USNRC to resolve these difficult and important issues. Current planning is that the program will continue through 1985. Expenditures for the fiscal year of 1980 are approximately $1.3 million dollars. The first phase effort scheduled to be completed at the end of Summer 1980 has two major objectives to be accomplished: (1) to determine the probability of a large LOCA (Loss of Coolant Accident) induced directly and indirectly by a range of earthquakes; (2) to develop a general methodology for combining generic dynamic loads for nuclear power plant design.

The objective of determining the probability of a large LOCA induced directly by earthquakes requires the development of a deterministic fracture mechanics model for fatigue crack growth. This model incorporated with stochastic inputs of initial crack size distribution, material properties, stress histories, and leak detection probability is used to calculate the probability of pipe leak and break when subjected to the various loading conditions including seismic. The best estimate model and approach were used in the computation. The probability results together with the uncertainties will be used as a basis to determine whether the low probability events such as a large LOCA and SSE should be combined when nuclear structures and components are designed. This model can also be used to evaluate the reliability of piping systems in general. The results of the evaluation will be used to provide technical basis for the USNRC to modify Regulatory Guide 1.46 (Protection Against Pipe Whip Inside Containment.

The objective of evaluating the probability of a large LOCA induced indirectly by earthquakes requires the identification of accident scenarios induced by earthquakes, such as mechanical and electric failures, explosions, missiles, and fires which, in turn, can cause a LOCA. For each scenario the probability is estimated.

The objective of developing a general methodology for combining generic dynamic loads involves the development of component reliability methodology to derive the appropriate load combination requirements for the nuclear power plant design. This methodology, by considering the event probability, nature of the events, structural properties, response characteristics, and other random phenomena, will achieve consistency in component design. This methodology and the developed model can also be used to evaluate the reliability of the existing structure and component design.

Due to the potential impact on licensing decisions, the NRC and the ACRS have requested that a National Peer Review Panel be formed to review our results and conclusions as well as technical detail. The Review Panel is comprised of five experts in the areas of metallurgy and materials, risk and reliability, structural and mechanical analysis, mathematical analysis, and nuclear system. Two review meetings have been scheduled prior to the ACRS meeting in November 1980.

This presentation describes the overview of the Load Combination Program. The consecutive presentations will address the technical approach, evaluation results and applications in more detail.

OVERVIEW OF PRECENTATION

- o INTRODUCTION
- o PROGRAM OVERVIEW
- o PROJECT I - LOCA-SSE DECOUPLING
- o PROJECT II - LOAD COMBINATION METHODOLOGY DEVELOPMENT
- o TECHNICAL PRESENTATION AGENDA

LOAD COMBINATION PROBLEM OVERVIEW

- o HOW TO DEFINE EACH INDIVIDUAL LOAD?
- o HOW TO DETERMINE THE LOAD COMBINATION EQUATIONS FOR NUCLEAR POWER PLANT DESIGN?
  - HOW TO POSTULATE PIPE BREAK LOCATIONS?
  - WHAT LOADS NEED TO BE COMBINED?
  - IF CERTAIN LOADS MUST BE COMBINED, WHAT IS THE MOST APPROPRIATE METHOD TO COMBINE THEM?
  - HOW MANY LOAD COMBINATION EQUATIONS ARE NEEDED?
- o HOW TO DETERMINE THE APPROPRIATE SERVICE LEVELS, LOAD CATEGORIES, AND DESIGN LIMITS?

ONLY TWO MAJOR ISSUES WERE TO BE ASSESSED IN THE FIRST PHASE OF THE PROGRAM (COMPLETED AT THE END OF AUGUST, 1980)

o EVENT DECOUPLING

TO CALCULATE THE PROBABILITY OF A LARGE LOCA INDUCED BY A RANGE OF EARTHQUAKES FOR A SELECTED PWR PLANT.

- DETERMINE WHETHER OR NOT LOW PROBABILITY EVENTS SUCH AS THE LOADS POSTULATED FOR THE DOUBLE-ENDED GULLOTINE BREAK IN THE REACTOR COOLANT LOOP NEED TO BE COMBINED WITH SSE SEISMIC.

- DEVELOP PROBABILISTIC FRACTURE MECHANICS MODEL TO AID IN POSTULATING PIPE BREAK LOCATIONS.

o LOAD COMBINATION METHODOLOGY DEVELOPMENT

TO DEVELOP A METHODOLOGY, BASED ON COMPONENT AND SYSTEM RELIABILITY, WHICH CAN BE USED TO:

- EVALUATE THE RELIABILITY OF THE EXISTING DESIGN.

- MORE RATIONALLY DETERMINE

  - WHICH LOADS NEED TO BE COMBINED
  - HOW TO COMBINE THEM
  - APPROPRIATE SERVICE LEVELS AND DESIGN
  - ALLOWABLES

LOAD COMBINATION PROGRAM

PHASE I WORK - SCHEDULE SUMMARY

- PHASE I WORK AUTHORIZED MARCH, 1979.
- PHASE I WORK PLAN DEVELOPED - JULY, 1979.
- TECHNICAL WORK STARTED - JULY, 1979.
- PHASE I TECHNICAL WORK COMPLETION - AUGUST, 1980.
- PHASE I FINAL REPORT (DRAFT) - NOVEMBER, 1980.

LOAD COMBINATION PROGRAM

PHASE I WORK - BUDGET SUMMARY

- TOTAL PHASE I BUDGET - $1256K (MARCH, 1979 THRU SEPTEMBER, 1980).
  - $1056K FROM MECHANICAL ENGINEERING RESEARCH BRANCH.
  - $200K FROM METALLURGY AND MATERIALS RESEARCH BRANCH.
- DISTRIBUTION OF FUNDS
  - PROJECT I - EVENT DECOUPLING - 70%
  - PROJECT II - LOAD COMBINATION METHODOLOGY DEVELOPMENT - 30%

LOAD COMBINATION PROGRAM
PHASE I WORK REVIEW SCHEDULE

- o FIRST PEER REVIEW PANEL MEETING
  SEPTEMBER 18 & 19, 1980
- o NRC/DIVISION OF SYSTEM & RELIABILITY RESEARCH REVIEW (RELIABILITY OF PIPING SYSTEMS)
  SEPTEMBER 23, 1980
- o NRC/NRR REVIEW
  SEPTEMBER 30, 1980
- o SENIOR CONSULTANT REVIEW (EVENT DECOUPLING)
  OCTOBER 13, 1980
- o SECOND PEER REVIEW PANEL MEETING
  NOVEMBER 6 & 7, 1980
  LIVERMORE, CALIFORNIA
- o ACRS MEETING - SUBCOMMITTEE ON COMBINATION OF DYNMAIC LOADS.
  NOVEMBER 18 & 19, 1980
  JUPITER, FLORIDA

PROJECT I ACTIVITY

TO CALCULATE THE PROBABILITY OF LEAK OR DOUBLE-ENDED GUILLOTINE BREAK INDUCED DIRECTLY BY A RANGE OF EARTHQUAKES FOR THE PRIMARY LOOP PIPING (HOT LEG, COLD LEG AND CORSS OVER) FOR A SELECTED PWR PLANT - ZION NUCLEAR POWER PLANT UNIT #1.

ZION UNIT #1 IS A PWR-1 WESTINGHOUSE 4-LOOP PLANT LOCATED 40 MILES NORTH OF CHICAGO ON LAKE MICHIGAN SHORELINE.

OWNER	-	COMMONWEALTH EDISON
A/E	-	SARGENT AND LUNDY
NSSS VENDER	-	WESTINGHOUSE
COMMERCIAL OPERATION	-	JUNE 1973
NET MWe	-	1100

WHY NRC WOULD LIKE TO CONSIDER DECOUPLING A LARGE LOCA AND SSE LOADS

- o THE ORIGINAL COMBINATION REQUIREMENT HAS NO RATIONAL BASIS.
- o THE COST OF PLANTS TO DESIGN FOR COMBINED LARGE LOCA AND SSE IS GREAT.
- o SOME OLD PLANTS WERE SIMPLY NOT DESIGNED FOR LOCA & SSE. MODIFICATION IS EITHER EXPENSIVE OR IMPOSSIBLE.
- o DESIGN FOR BOTH LOCA & SSE MAY NOT BE BENEFICIAL BECAUSE IT MAY REDUCE SYSTEM RELIABILITY DURING NORMAL OPERATION.
  - A LARGE LOCA IN THIS EVALUATION IS DEFINED AS A DOUBLE-ENDED GUILLOTINE BREAK IN THE PRIMARY LOOP PIPING SYSTEM, I.E., HOT LEG, COLD LEG, AND CROSS OVER.
  - EVEN IF THE DECOUPLING DECISION IS APPLIED ONLY TO PRIMARY LOOP PIPING FRACTURE INSIDE THE REACTOR CAVITY WALL OF PWR PLANTS IT WILL BE OF GREAT BENEFIT TO THE NUCLEAR INDUSTRY.

TECHNICAL PRESENTATION WILL BE GIVEN IN DETAIL

- o FRACTURE MECHANICS MODEL
- o INPUT VARIABLES
- o FAILURE CRITERION
- o PROBABILISTIC COMPUTATION
- o ADVANCES IN PIPING RELIABILITY MODEL
- o POSSIBLE APPLICATION OF THE DEVELOPED PIPING RELIABILITY MODEL
- o RESULTS AND CONCLUSIONS

PROJECT II ACTIVITY

TO DEVELOP A METHODOLOGY BASED ON COMPONENT AND SYSTEM RELIABILITY WHICH CAN BE USED TO:

- o EVALUATE THE RELIABILITY OF THE EXISTING DESIGN
- o MORE RATIONALLY DETERMINE
  - WHICH LOADS NEED TO BE COMBINED
  - HOW TO COMBINE THEM
  - APPROPRIATE SERVICE LEVELS AND DESIGN ALLOWABLES.

EXAMPLES OF LOAD COMBINATION REQUIREMENT

STEEL STRUCTURE DESIGN (SRP 3.8.4)

$$0.9Y \geq D+L+T_A+R_A+1.25P_A+1.0(Y_J+Y_R+Y_M)+1.25E$$

SPECIFIED BY NRC

MECHANICAL PIPING DESIGN

$$2.25\ S_M \geq OBE + CO + CH + P + D$$

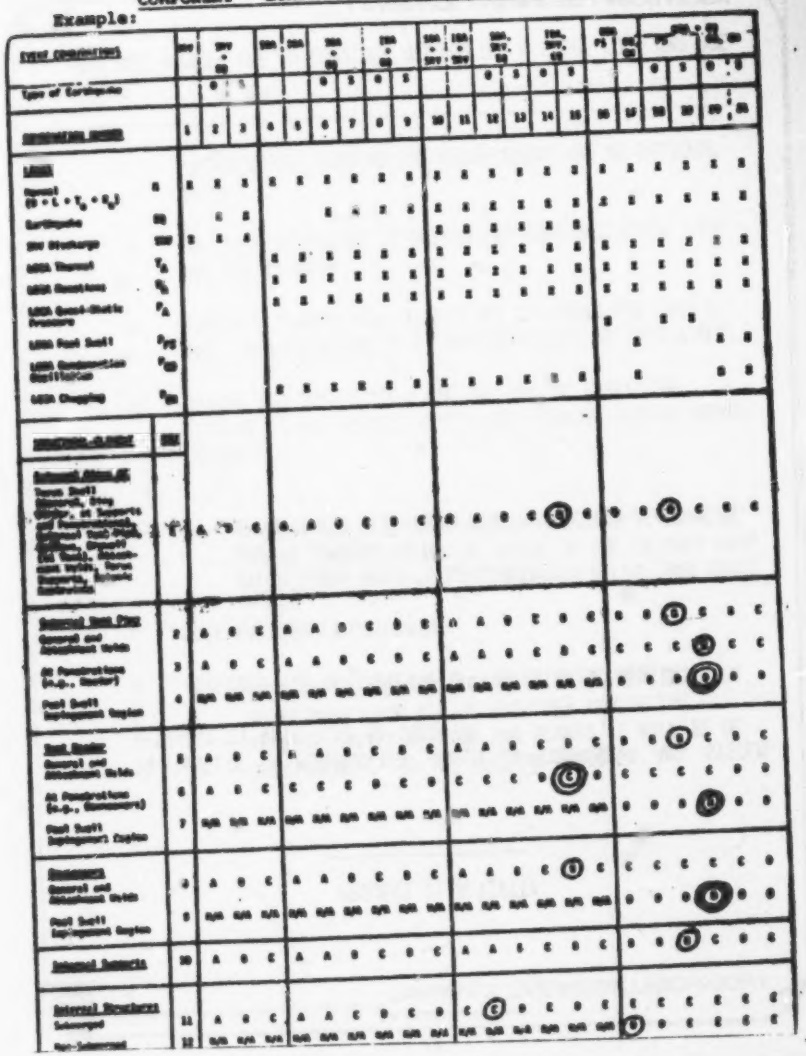
SPECIFIED BY ASME CODE

SPECIFIED BY OWNERS AND THEIR A/E'S APPROVED BY NRC

MARK I CONTAINMENT

COMPONENT - LOADINGS - SERVICE LEVEL ASSIGNMENTS

Example:



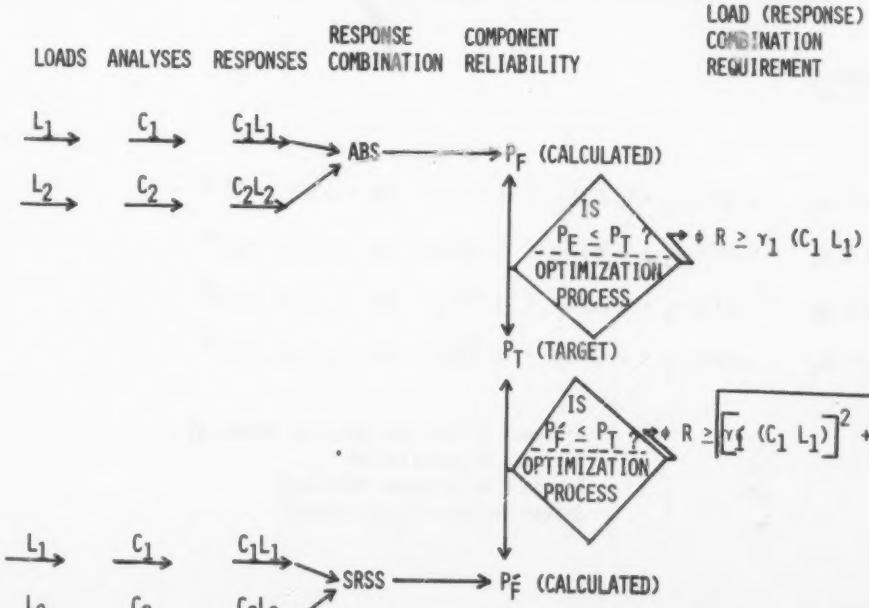
CONCEPT OF USING COMPONENT RELIABILITY METHODOLOGY TO EVALUATE EXISTING DESIGN

LOADS	ANALYSES	RESPONSES	RESPONSE COMBINATION	COMPONENT RELIABILITY
L_1	C_1	C_1L_1	$\phi R \geq C_1L_1 + C_2L_2$	P_F (ABS)
L_2	C_2	C_2L_2		
			COMPARE WITH CRITERIA FOR ACCEPTANCE	
L_1	C_1	C_1L_1	$\phi R \geq \sqrt{(C_1L_1)^2 + (C_2L_2)^2}$	P_F (SSRS)
L_2	C_2	C_2L_2		

LAWRENCE LIVERMORE LABORATORY
University of California

17

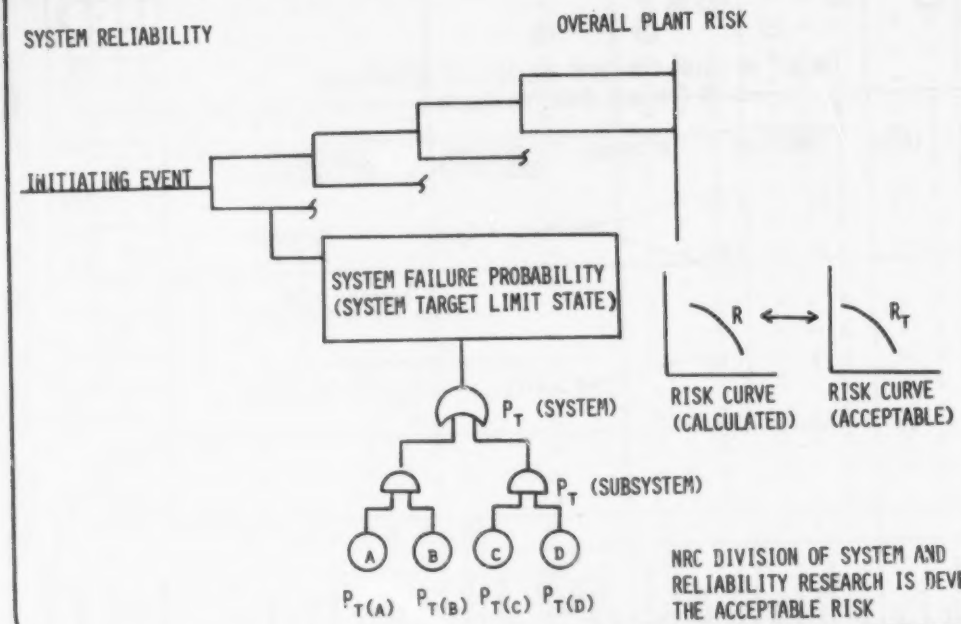
CONCEPT OF USING COMPONENT RELIABILITY METHODOLOGY TO DEVELOP LOAD (RESPONSE) COMBINATION REQUIREMENTS



LAWRENCE LIVERMORE LABORATORY
University of California

18

CONCEPT OF USING SYSTEMS RELIABILITY METHODOLOGY TO DEVELOP COMPONENT TARGET LIMIT STATE



NRC DIVISION OF SYSTEM AND RELIABILITY RESEARCH IS DEVELOPING THE ACCEPTABLE RISK

LAWRENCE LIVERMORE LABORATORY
University of California

19

THE DESIGNER IS PROVIDED WITH A SERIES OF DETERMINISTIC FACTORED DESIGN EQUATIONS WHICH WILL ENSURE THE ACHIEVEMENT OF A TARGET LIMIT STATE PROBABILITY

EXAMPLE

$$1.5\ s_M = 0.875c_1v_0 + 0.873c_2L + 1.034c_3P \quad \text{FOR T.L.S.P.} = 10^{-2}$$

$$1.5\ s_M = 0.957c_1v_0 + 0.951c_2L + 1.197c_3P \quad \text{FOR T.L.S.P.} = 10^{-4}$$

$$1.5\ s_M = 1.033c_1v_0 + 1.023c_2L + 1.324c_3P \quad \text{FOR T.L.S.P.} = 10^{-6}$$

$$1.5\ s_M = 1.108c_1v_0 + 1.076c_2L + 1.434c_3P \quad \text{FOR T.L.S.P.} = 10^{-8}$$

c_1v_0 = RESPONSE TO INITIAL VELOCITY TRANSIENT
c_2L = RESPONSE TO DEAD WEIGHT
c_3P = RESPONSE TO INTERNAL PRESSURE
s_M = STRESS INTENSITY LIMIT (CODE)

LAWRENCE LIVERMORE LABORATORY
University of California

20

LAWRENCE LIVERMORE NATIONAL LABORATORY

LOAD COMBINATION PROGRAM

PROJECT I
EVENT DECOUPLING

PRESENTED BY

S. C. LU & R. D. STREIT

GAITHERSBURG, MARYLAND

OCTOBER 31, 1980

LAWRENCE LIVERMORE LABORATORY
University of California

Probabilistic Assessment of Decoupling The Effects of Large LOCA and Earthquake in Nuclear Power Plant Design*

S. C. Lu and R. D. Streit
University of California
Lawrence Livermore National Laboratory
Livermore, California 94550

Structures, systems, and components important to the safety of nuclear power plants in the United States have been historically designed to withstand the combined effects of an earthquake and a loss of coolant accident (LOCA). In the past, the LOCA-earthquake combination has been treated with a conservative "worst case" philosophy in order to simplify the design procedure. This approach has resulted in great penalty in terms of both cost and design practicality. The results have been fast increased cost for new plants and expensive or unfeasible modifications for existing plants subjected to safety reevaluation. Additionally, there has been an increasing concern whether design for extreme loads will result in reduced reliability for the normal plant operation. It finally has become a strong desire of the nuclear industry to decouple the effects of LOCA-earthquake combination.

A research project has been conducted at Lawrence Livermore National Laboratory to establish a rational basis for the LOCA-earthquake decoupling decision. It was determined that the decoupling decision should be based on probability, therefore a systematic probability approach has been employed to treat realistically the many random elements considered in the assessment. Since the probability of simultaneous occurrence of an independent LOCA and an earthquake is extremely low this assessment considered only LOCAS which are dependent on, or directly and indirectly induced by, earthquakes.

Activities in Phase I of this project have focused on the probability of a large LOCA induced by an earthquake for a pressurized water reactor (PWR) plant. A large LOCA is defined as a double-ended guillotine break of the primary reactor coolant loop (RCL) piping, i.e., the hot leg, cold leg, and crossover leg. Zion unit 1 power station was selected for the evaluation. Zion-1, owned and operated by Commonwealth Edison Company (SECO), utilizes a four-loop Westinghouse nuclear steam supply system (NSSS).

*Presented at the 8th Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, October 27-31.

The probability of a LOCA directly induced by earthquakes was assessed by a numerical simulation of pipe rupture of a reactor coolant system. The probabilistical approach of the simulation employed a deterministic fracture mechanics model which dictates the fatigue growth of pre-existing cracks in the pipe. The simulation procedure accounts for the stochastic nature of input elements such as the initial crack size distribution, the crack occurrence rate, crack and leak detection probabilities as functions of crack size, plant transient occurrence rates, the seismic hazard curve, stress histories, and crack growth model parameters. Effects on final results due to variations and uncertainties of input elements were assessed by a sensitivity study.

Analyses were performed to evaluate stress histories in the primary coolant loop piping of Zion-1 nuclear power station. The results of the analyses were used as input to a simulation procedure for predicting pipe rupture probabilities of the reactor coolant system. In order to make the simulation as realistic as possible, best estimate rather than conservative assumptions were considered throughout. The best estimate solution, however, was aided by a sensitivity study to assess the possible variation of outcomes resulted from uncertainties associated with these assumptions. Sources of stresses considered in the evaluation were pressure, dead weight, restraint of thermal expansion, thermal gradients through the pipe wall, residual welding, pump vibrations, and finally seismic excitations.

The fracture mechanics model considered the initial length and depth distribution of semi-elliptical cracks characterized by a bivariate distribution. It was assumed that the crack length must be greater than or equal to the crack depth. Once the crack size characteristics were determined, the model was then adjusted to the probability that a flaw exists and is not detected. The sensitivity of the piping reliability due to various crack size distributions was considered in the analysis.

The fatigue model for crack growth employed a Paris-type growth rate equation modified for the mean stress level. The semi-elliptical shape of the fatigue crack was assumed to be maintained during crack growth. The crack growth rate, governed by weighted stress intensity solutions for the depth and length, respectively was evaluated for each class of transients.

The weighted stress intensity solutions were used in the fatigue growth model to extend the cracks, thereby changing the crack size distribution as a function of time. Critical crack sizes for pipe leaks and LOCA were evaluated using elastic and elastic-plastic concepts, respectively. The leak rate was calculated using linear elastic crack opening displacement analysis; whereas, net section plastic instability was found to

be the critical criteria for a LOCA. Since the leak resulting from the minimum through-wall crack (2 times the wall thickness) was large, i.e., greater than 2 gpm, all leaks were considered large. In the current study their detection probability was taken to be one.

The numerical simulation of the life history of a reactor primary coolant system included all random elements which dictate a possible pipe rupture. These elements are crack initiation and distribution, pre-service and in-service inspections, loads and stresses, and leak detection. A computer code was developed to randomly simulate life history replications. Each weld joint was simulated through all replications.

Inputs were the crack occurrence rate, weld joint volumes, crack length and depth distribution, crack and leak detection probabilities as functions of crack size, transient occurrence rates, seismic hazard curve, transient and earthquake stresses, and crack growth model parameters. Input parameter estimates were obtained from published reports and from operating data on Zion-1.

Simulation outputs were times to LOCAs in all weld joints, the causes, and the number of replications. The probability of primary coolant loop pipe rupture was estimated to be the sum of the failure probabilities of all joints. This is accurate if all probabilities are small and joint failures are independent events.

Results of the simulation indicated that the probability of a large LOCA either in the absence of or including earthquakes was very small. The unconditional probability of a large LOCA and no earthquake occurrence in the life of the plant was estimated in the neighborhood of 10^{-12}. Seismic stresses tended to increase the LOCA probability but only to an almost negligible extent. The unconditional probability of a large LOCA and an earthquake in the plant life was about 5 x 10^{-14}. The probability of a leak was found to be several magnitudes greater than that of a complete pipe rupture, clearly supporting the "leak before break" philosophy.

A LOCA large or small, can be indirectly induced by earthquakes because of an explosion, a fire, an electrical failure, or a missile generated from the failure of a structure or a mechanical component during an earthquake. Scenarios for potential sources were constructed and analyzed. Engineering judgements, rather than more exact analyses, were exercised to order these scenarios probabilistically. It was discovered that most of the LOCAs indirectly induced by earthquakes did not fall within the definition of a large LOCA. The probability of a double-ended guillotine break, or a large LOCA, of the reactor coolant loop piping as an indirect result of an earthquake was again found to be small, i.e., less than 10-10 for a given SSE.

LAWRENCE LIVERMORE NATIONAL LABORATORY

LOAD COMBINATION PROGRAM

PROJECT I
EVENT DECOUPLING

OVERVIEW

S. C. LU

8TH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING

GAITHERSBURG, MARYLAND

OCTOBER 31, 1980

OBJECTIVE IS TO PROVIDE TECHNICAL JUSTIFICATIONS FOR DETERMINING WHETHER OR NOT TO DECOUPLE CERTAIN LOW-PROBABILITY EVENT COMBINATIONS

- o 10CFR50 IS THE BASIS FOR COMBINATION
- o CURRENT PRACTICE OF EVENT COMBINATION IS OVERLY CONSERVATIVE AND MAY NOT BE BENEFICIAL FOR DAY-TO-DAY PLANT OPERATION.
- o EXAMPLES OF LOW-PROBABILITY EVENT COMBINATIONS:
  - EQ + LOCA
  - EQ + WATER HAMMER
  - EQ + MISSILES
  - EQ + PIPE WHIP
  - OTHERS
- o CURRENT OBJECTIVE IS TO ESTABLISH A RATIONAL BASIS FOR LOCA-EQ DECOUPLING DECISION.

DECOUPLING LOCA AND SSE IS DESIRABLE, WHY?

- o THE COST OF PLANTS TO DESIGN FOR COMBINED LOCA AND SSE IS GREAT.
- o SOME OLD PLANTS WERE SIMPLY NOT DESIGNED FOR LOCA + SSE. MODIFICATION IS EITHER EXPENSIVE OR IMPOSSIBLE.
- o DESIGN FOR LOCA + SSE MAY NOT BE BENEFICIAL TO PLANTS BECAUSE OF REDUCED SYSTEM RELIABILITY FOR NORMAL OPERATIONS.

OUR DECOUPLING DECISION IS BASED ON PROBABILISTIC ASSESSMENT

- NATURAL AND ACCIDENTAL EVENTS AND MATERIAL PROPERTIES ARE RANDOM PROCESSES. A PROBABILISTIC APPROACH IS BEST FOR DECISION MAKING INVOLVING RANDOM PROCESSES.
- A PROBABILISTIC APPROACH CAN MORE ACCURATELY REPRESENT REALISTIC SITUATIONS AND ELIMINATES UNDESIRABLE AND INCONSISTENT CONSERVATISM INTRODUCED BY A DETERMINISTIC METHODOLOGY.
- DETERMINISTIC SAFETY MARGIN DOES NOT REPRESENT PROBABILITY OF FAILURE UNDER RANDOM LOADING.

PHASE I ACTIVITIES FOCUS ON LARGE LOCA FOR ZION UNIT I (A PWR PLANT)

- DOUBLE-ENDED PIPE FRACTURE OF RCL PIPING SYSTEM.
- LOCA'S DIRECTLY INDUCED BY EARTHQUAKE.
  - DEFINE LOADS AND COMPUTE STRESSES
  - PERFORM PROBABILISTIC PIPE FAILURE ANALYSIS TO PREDICT PIPE RUPTURE PROBABILITY
  - THE DEVELOPED METHODOLOGY CAN BE APPLIED TO BRANCH LINES, SECONDARY PIPING SYSTEM, OTHER PWR AND BWR PLANTS FOR DECOUPLING DECISION AND PIPING RELIABILITY ASSESSMENT.
- LOCA'S INDIRECTLY INDUCED BY EARTHQUAKE
  - IDENTIFY SOURCES: STRUCTURAL, MECHANICAL, AND ELECTRICAL FAILURES, EXPLOSIONS, FIRES, MISSILES, AND OTHERS.
  - CONSTRUCT SCENARIOS
  - ESTIMATE PROBABILITY (ENGINEERING JUDGEMENT)
- COST AND SAFETY MARGIN ASSESSMENT DUE TO LOCA + SSE REQUIREMENT.

11-140

LAWRENCE LIVERMORE NATIONAL LABORATORY

LOAD COMBINATION PROGRAM

PROJECT I
EVENT DECOUPLING

LOADS AND STRESSES

S. C. LU

8TH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING

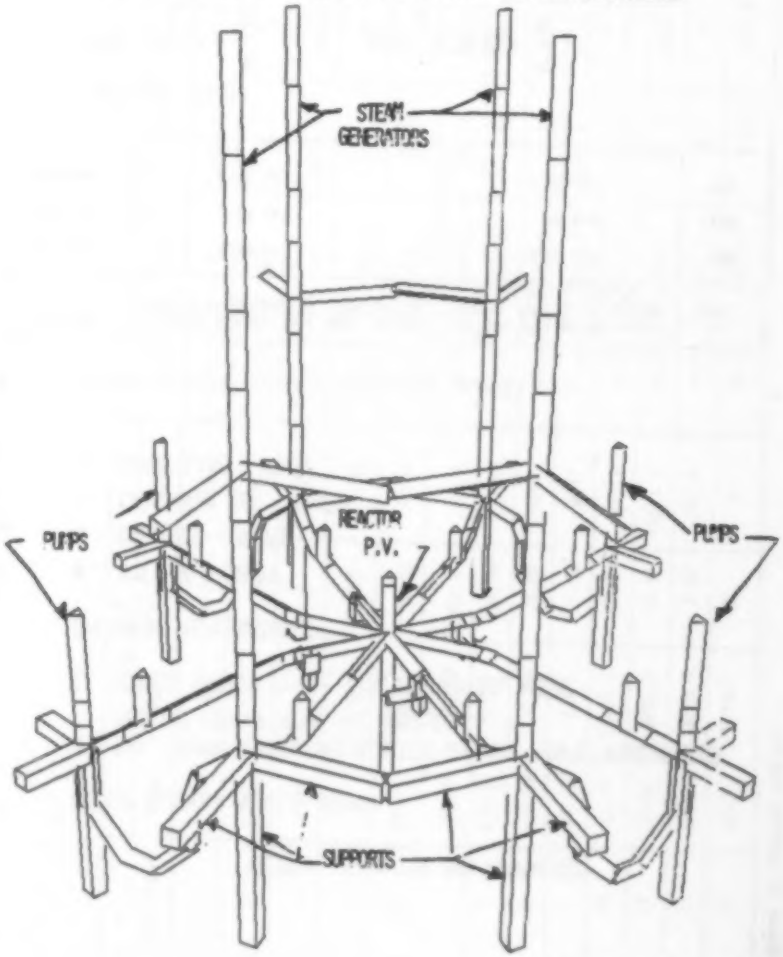
GAITHERSBURG, MARYLONG

OCTOBER 31, 1980

STRESS CALCULATIONS ARE BASED ON ADVANCED METHODS AND MODELING TECHNIQUES

- ANALYTICAL SOLUTIONS FOR PRESSURE AND THERMAL GRADIENT STRESSES.
- FINITE ELEMENT METHOD FOR WEIGHT AND THERMAL EXPANSION STRESSES
- SEISMIC STRESSES WERE EVALUATED BY TIME HISTORY ANALYSES WHICH ACCOUNT FOR SOIL STRUCTURE INTERACTION, COUPLING EFFECT BETWEEN BUILDING STRUCTURE AND PIPING SYSTEM.
- UNCERTAINTIES WERE INVESTIGATED IN A SENSITIVITY STUDY.

THE NSSS MODELS ARE DETAILED ENOUGH TO PRODUCE ACCURATE STRESSES



TRANSIENT STRESSES WERE CALCULATED USING MEASURED TEMPERATURES
RES FROM ZION I WHERE POSSIBLE

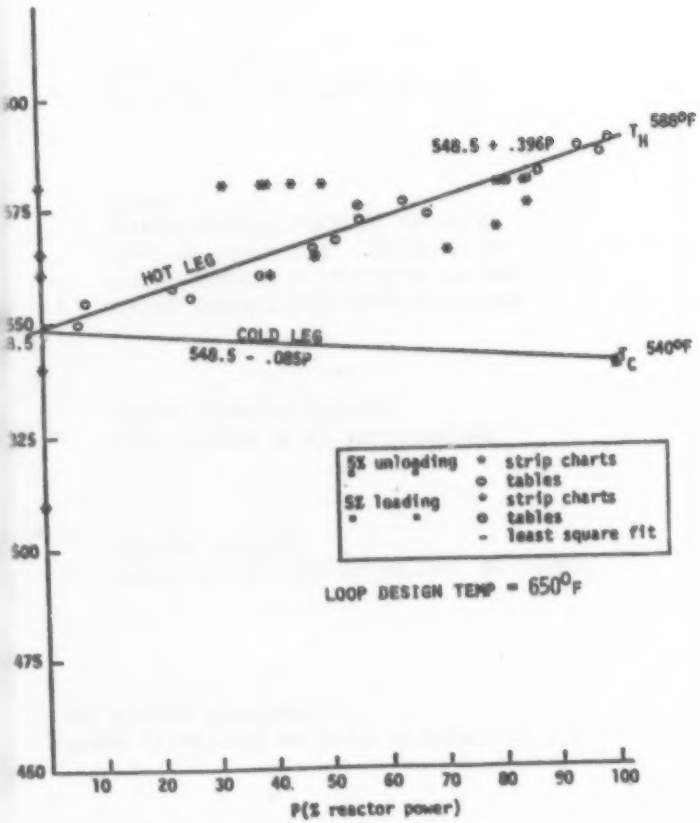


Figure 1. Loading and unloading temperatures.

THERMAL EXPANSION AND DEADWEIGHT

FINITE ELEMENT MODEL - SAPIV

- RVP, STEAM GENERATORS, REACTOR COOLANT PUMPS, PRESSURIZER, AND THE LOOP PIPING ARE MODELED
- STRESS OUTPUT AT ALL CIRCUMFERENTIAL WELDS

OPERATING TEMPERATURES

- HOT LEG - 588°F
- COLD LEG - 540°F
- CROSSOVER LEG - 540°F
- SURGE LINE - 620°F

STRESS RESULTS (MAXIMUM STRESS IN LOOP 1)

LOCATION	THERMAL EXPANSION + DEAD WEIGHT STRESS AT OPERATING TEMPERATURE	THERMAL EXPANSION + DEAD WEIGHT STRESS AT DESIGN TEMPERATURE (650°F)	% DIFF.
HOT LEG	8.54 ksi	9.75 ksi	14%
COLD LEG	.73 ksi	.64 ksi	-12%
CROSSOVER	2.78 ksi	3.26 ksi	17%

PRESSURE STRESS

$$\text{HOOP STRESS} = \frac{PD}{2T} \qquad \text{LONG. STRESS} = \frac{PD}{4T}$$

TYPE OF TRANSIENTS

ents No.	Description
1	Reactor Heatup
2	Reactor Cool Down
3	Plant Loading, 5% Per Minute
4	Plant Unloading, 5% Per Minute
5	Ten Percent Step Load Decrease
6	Ten Percent Step Load Increase
7	Large Step Decrease in Load
8	Loss of Load From Full Power
9	Loss of Offsite Power
10	Loss of Flow in One Loop, the Last Loop
11	Loss of Flow in One Loop, the Other Loops
12	Reactor Trip From Full Power
13	Steam Line Break From No Load

17

PIPE SECTION MAXIMUM AND MINIMUM
INSIDE SURFACE RADIAL GRADIENT STRESS

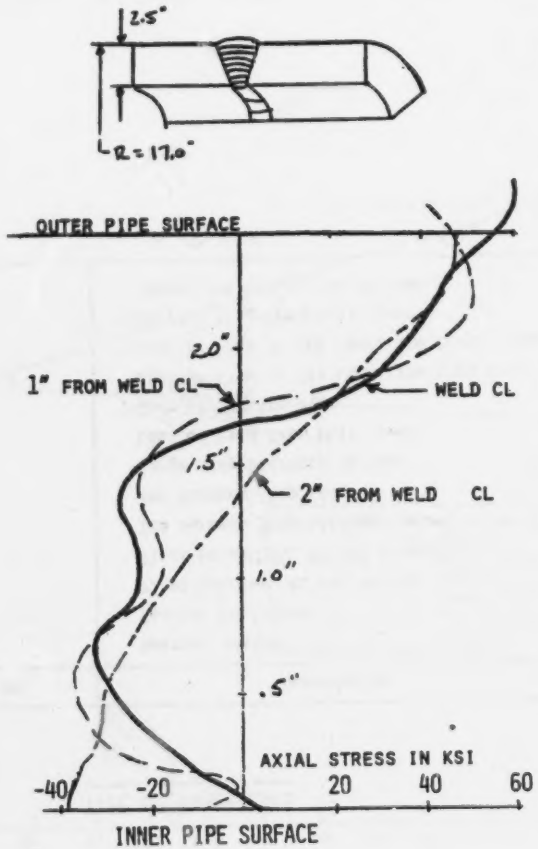
Transient	Cold Leg				Hot Leg			
	Time	Max.	Time	Min.	Time	Max.	Time	Min.
1	0.	0.	17998	-2737.	0	0	17953.3	-3410
2	14389	2770	0	0	17865.9	3450	0	0
3	1020	783	0	0	0	0	1019.7	-4327.4
4	0	0	1020	-783	1019.8	4342.9	0	0
5	250.1	1322	50.06	-4032	300.0	2212.1	29.77	-1528.6
6	49.98	4033.2	1321.9	250.2	29.50	1648.9	299.96	-2101.7
7	840.02	2136	36.001	-3435	820.0	9292.6	60.07	-1456.8
8	100.05	735.8	26.87	-10463	100.07	18103.8	26.9	-8023.7
9	140.41	633	5.68	-3794	150.0	6160.2	2250.0	-2825.8
10	13.67	13806.6	42.27	-981.7	45.07	34605.6	15.93	-1559.0
11	100.01	3296.3	15.91	-5071.3	100.98	14308.5	4.12	-4587.0
12	39.97	3020.0	6.99	-942.9	39.61	21317.5	0	0
13	50.1	74118.7	0.	0.	60.01	74870.0	4.99	-940.6

Units in seconds and psi

18

pages tightly bound

WELDING PRODUCES EITHER COMPRESSIVE OR VERY SMALL TENSILE RESIDUAL STRESSES ON THE INSIDE SURFACE OF THE PIPE



VIBRATIONAL STRESS

DYNAMIC ANALYSES:

1. RESPONSE SPECTRUM
2. TIME HISTORY WITH HARMONIC VIBRATING MOTION AT PUMP

EXPERIMENT DATA:

TEST REPORT FROM INDIAN POINT UNIT 2 (WCAP-7920)

MAXIMUM DEFLECTION AT THE TOP OF STEAM GENERATOR

TANGENTIAL - 7×10^{-3} IN
RADIAL - 4×10^{-3} IN

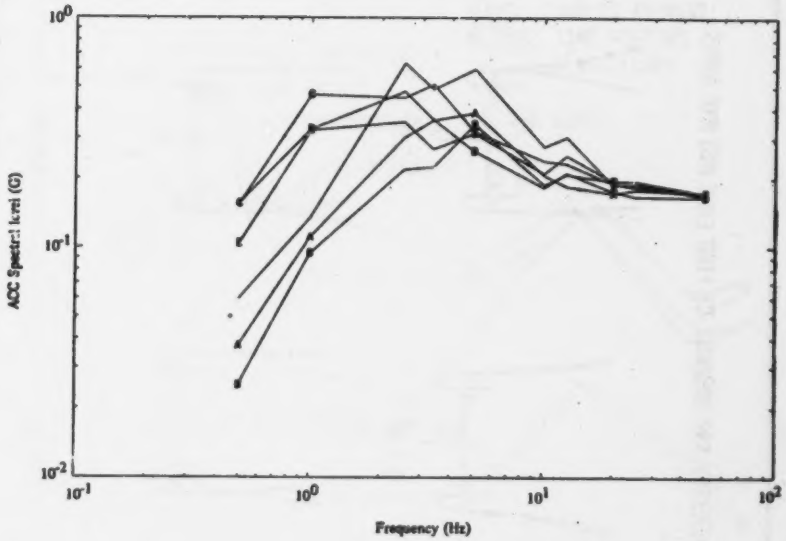
READER FICHE
best copy available
pages tightly bound

STRESSES AND DAMAGE FACTORS WERE PRODUCED AT VARIOUS EARTHQUAKE INTENSITIES

INTENSITY	NO OF EQ. SIMULATED
OBE	1
SSE	4
3 SSE	4
5 SSE	1

OBE = OPERATING BASIS EARTHQUAKE

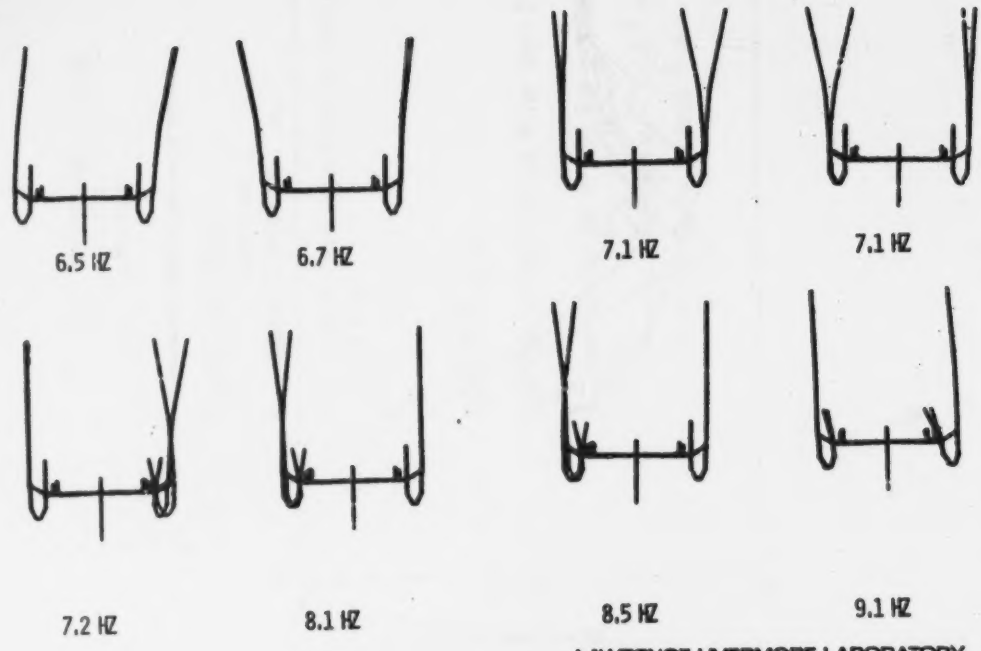
SSE = SAFE SHUTDOWN EARTHQUAKE
= .17 G ZPA OR PEAK ACCEL.



Acceleration spectra at SSE for 5% damping

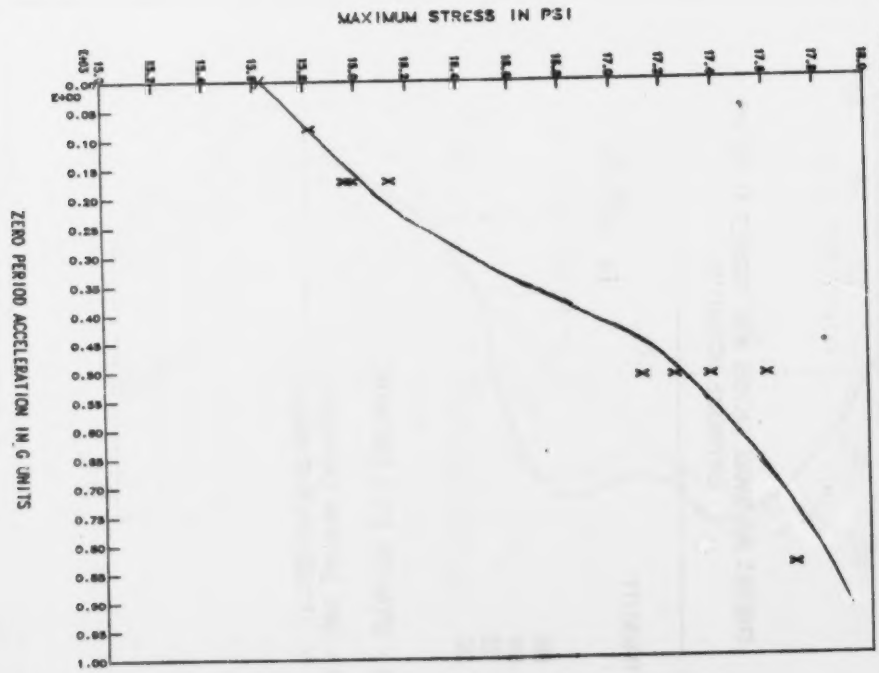
Figure A.4-1

STEAM GENERATOR SWAY DOMINATES THE FIRST EIGHT NSSS MODE SHAPES



LAWRENCE LIVERMORE LABORATORY
University of California

THE CONTRIBUTION OF THE SEISMIC MOTION TO THE MAXIMUM STRESS AT THE HOT LEG TO PRESSURE VESSEL JOINT IS SMALL

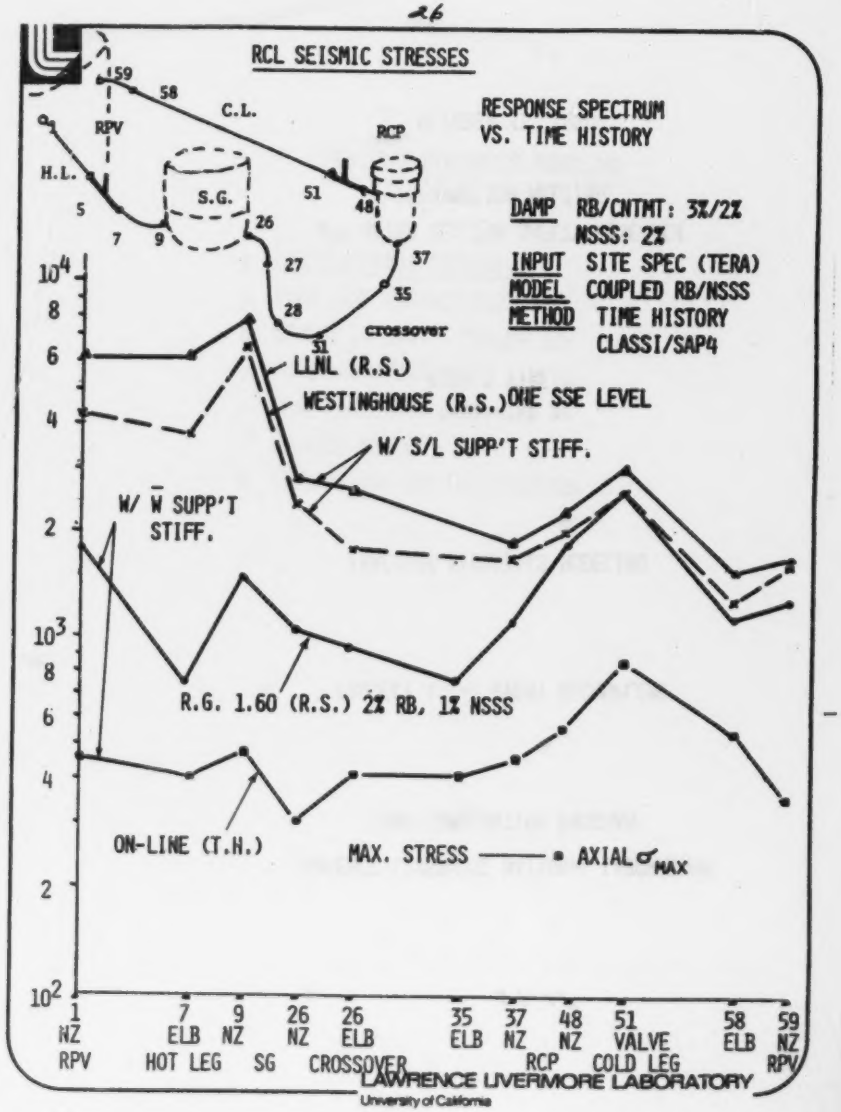


SENSITIVITY STUDY

PARAMETERS CONSIDERED:

- DAMPING
- SEISMIC INPUT TIME HISTORY
- SOIL-STRUCTURE INTERACTION
- RESPONSE SPECTRUM ENVELOPING VS. TIME HISTORY ANALYSIS

RCL SEISMIC STRESSES



27-28

LAWRENCE LIVERMORE NATIONAL LABORATORY

LOAD COMBINATION PROGRAM

PROJECT I - EVENT DECOUPLING

FRACTURE MECHANICS MODELING

PRESENTED BY

RONALD STREIT

8TH WATER REACTOR SAFETY RESEARCH

INFORMATION MEETING

OCTOBER 31, 1980

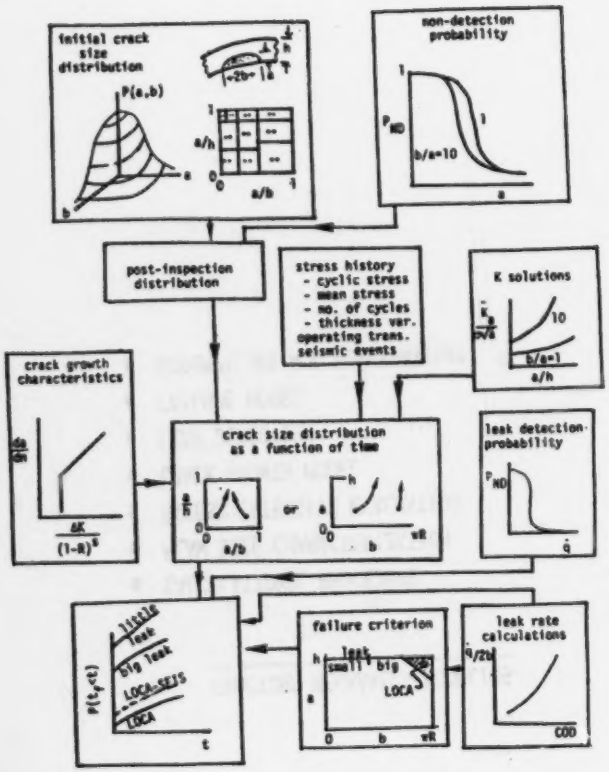
FRACTURE MECHANICS MODELING

- CALCULATIONAL PROCEDURE
- FLAW SIZE CHARACTERIZATION
- STRESS INTENSITY EVALUATION
- CRACK GROWTH MODEL
- LEAK DETECTION
- FAILURE MODEL
- ASSUMPTIONS AND OBSERVATIONS

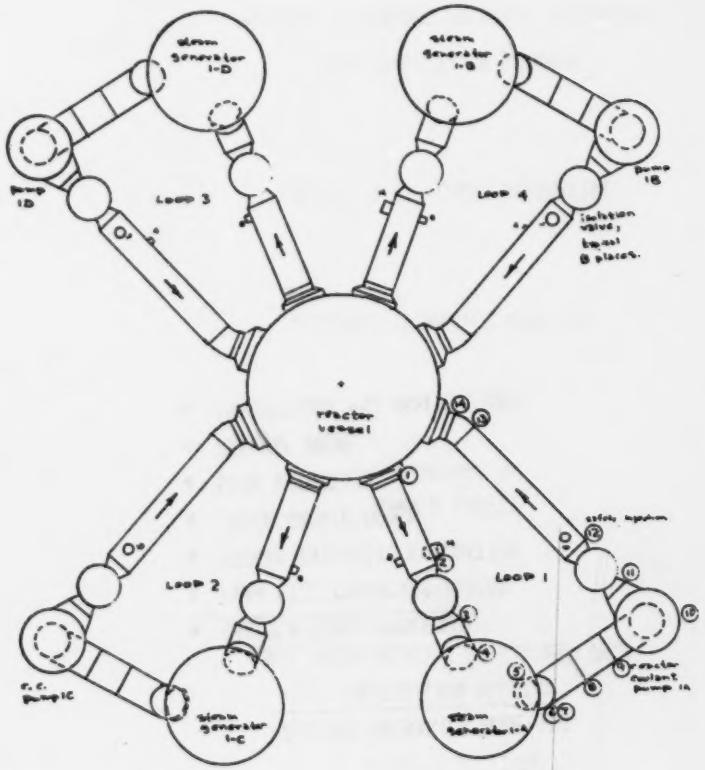
FRACTURE MECHANICS MODELING

- CALCULATIONAL PROCEDURE
- FLAW SIZE CHARACTERIZATION
- STRESS INTENSITY EVALUATION
- CRACK GROWTH MODEL
- LEAK DETECTION
- FAILURE MODEL
- ASSUMPTIONS AND OBSERVATIONS

CALCULATIONAL PROCEDURE



PRIMARY COOLANT LOOP GIRTH WELDS ARE EVALUATED IN THE CURRENT STUDY



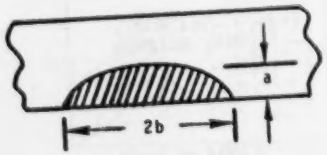
REACTOR COOLANT SYSTEM DESIGN TRANSIENTS

	DESIGN FREQUENCY (PER YEAR)	ACTUAL ZION I FREQUENCY
HEATUP AND COOLDOWN	5 (EACH)	6.74 (EACH)
UNIT LOADING AND UNLOADING (5% OF FULL POWER/MIN)	457 (EACH)	18.0 (LOADING) 13.3 (UNLOADING)
STEP LOAD INCREASE AND DECREASE (10% OF FULL POWER)	50 (EACH)	13.2
LARGE STEP LOAD DECREASE (WITH STEAM DUMP)	5	1.7
LOSS OF LOAD	2	--
LOSS OF POWER	1	--
LOSS OF FLOW	2	--
REACTOR TRIP (FROM FULL POWER)	10	22.2
INADVERTENT AUXILIARY SPRAY	.25	--
STEAM LINE BREAK	1/40	--
SAFETY INJECTIONS	--	3.9

FRACTURE MECHANICS MODELING

- CALCULATIONAL PROCEDURE
- FLAW SIZE CHARACTERIZATION
- STRESS INTENSITY EVALUATION
- CRACK GROWTH MODEL
- LEAK DETECTION
- FAILURE MODEL
- ASSUMPTIONS AND OBSERVATIONS

INITIAL CRACK SIZE DISTRIBUTION



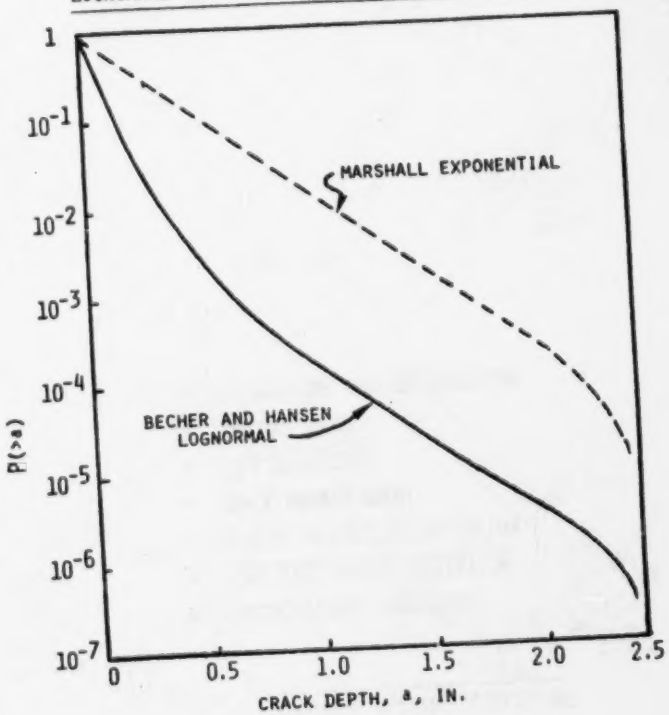
DEFINED BY

- a
- b/a
- PROB. OF HAVING A CRACK IN A UNIT VOLUME

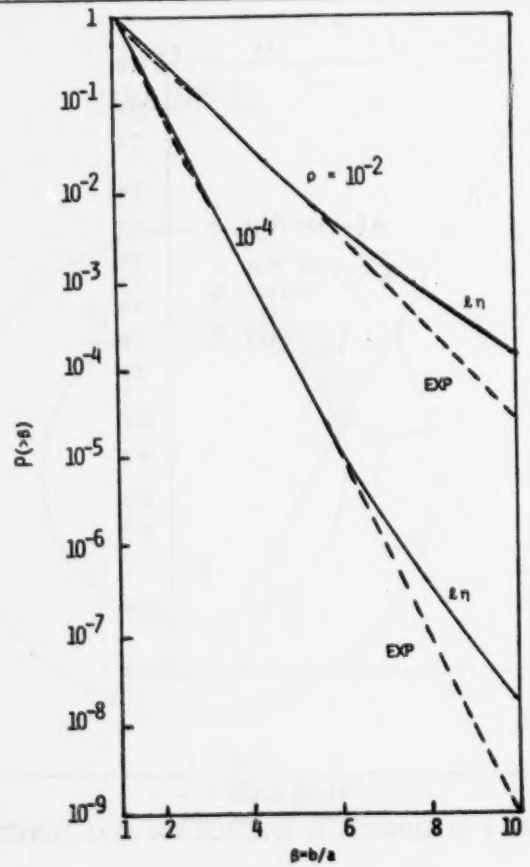
ASSUME: $\frac{b}{a}$ NOT CORRELATED TO a.

$p(a)$, $p(\frac{b}{a})$, p_v* THEN DEFINE INITIAL CRACK SIZES AND PROBABILITIES.

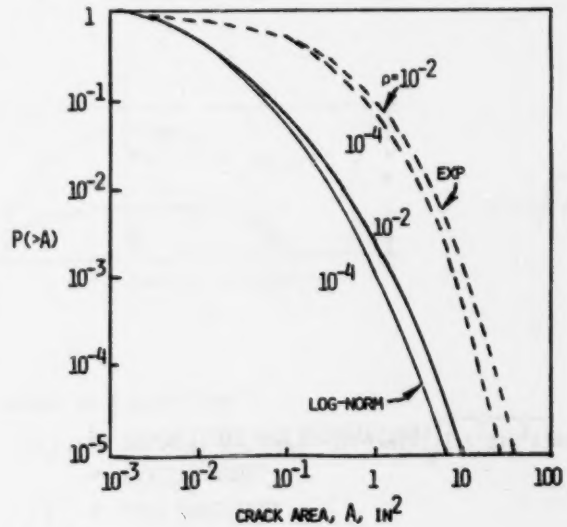
CRACK SIZE DISTRIBUTIONS HAVE BEEN ESTIMATED USING BOTH LOGNORMAL AND EXPONENTIAL DISTRIBUTIONS



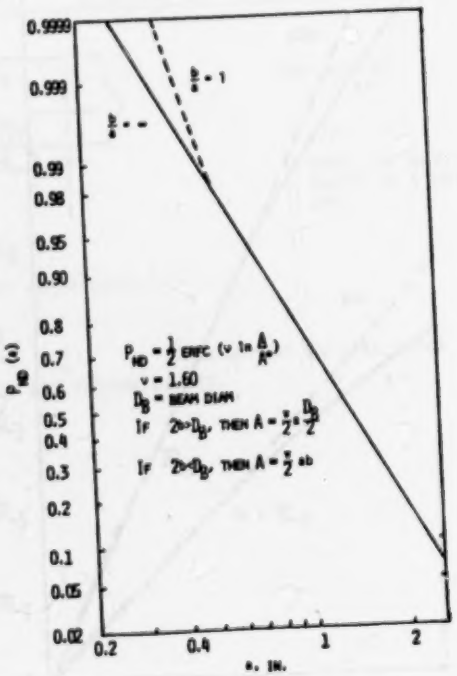
THE ASPECT RATIO, B/A, IS PREDICTED BY FIXING A POINT ON THE DISTRIBUTION



THE CRACK AREA CAN THEN BE EVALUATED BASED ON THE PIPE DIMENSIONS AND THE DISTRIBUTIONS FOR A AND B/A



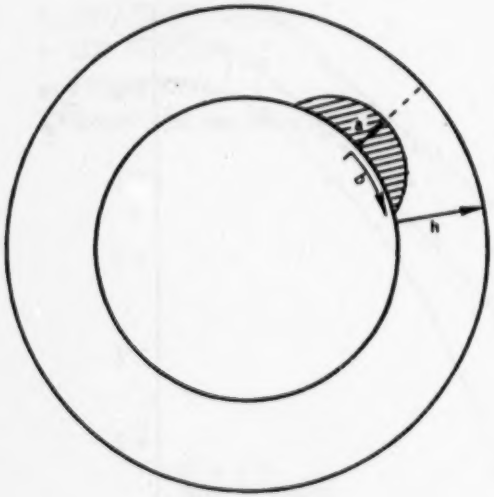
PROBABILITY OF NON-DETECTION IS ESTIMATED AS A FUNCTION OF THE CRACK DEPTH



FRACTURE MECHANICS MODELING

- CALCULATIONAL PROCEDURE
- FLAW SIZE CHARACTERIZATION
- STRESS INTENSITY EVALUATION
- CRACK GROWTH MODEL
- LEAK DETECTION
- FAILURE MODEL
- ASSUMPTIONS AND OBSERVATIONS

CRACK GEOMETRY

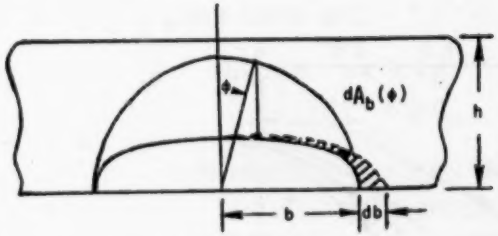


$\alpha = a/h$

$\ell = 2b$

ℓ/a = ASPECT RATIO

2 DEGREE OF FREEDOM
SEMI-ELLIPTICAL CRACK



$$k = \sqrt{1-\frac{a^2}{b^2}}$$

FOR GROWTH IN b DIRECTION

$$dA_b(\phi) = b\,db\ \sin\phi\ \sin\left[\tan^{-1}\left(\frac{a}{b}\tan\phi\right)\sqrt{1-k^2\sin^2\phi}\right]d\phi$$

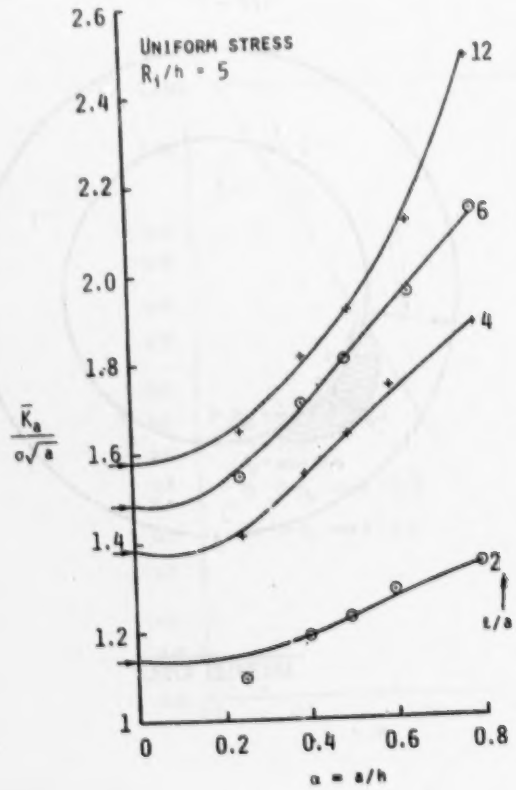
$$\bar{K}_b = \left\{\int_0^{\frac{\pi}{2}} K^2(\phi)\ dK_b(\phi)\right\}^{1/2}$$

$$\bar{K}_a = \left\{\int_0^{\frac{\pi}{2}} K^2(\phi)\ dA_a(\phi)\right\}^{1/2}$$

$$\frac{da}{dn} = C\,\bar{K}_a'^{\,m}$$

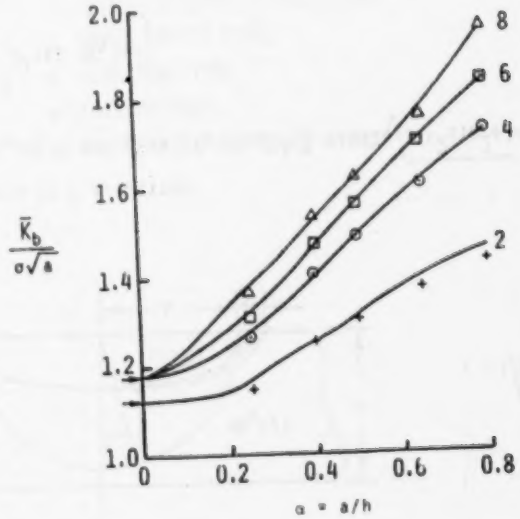
$$\frac{db}{dn} = C\,\bar{K}_b'^{\,m}$$

FOR A GIVEN TRANSIENT, IE., STRESS DISTRIBUTION, THE STRESS INTENSITY IS CALCULATED FOR VARIOUS CRACK GEOMETRIES



o FOR THE CRACK DEPTH.....

.....AND SIMILARLY, FOR THE CRACK LENGTH



FRACTURE MECHANICS MODELING

- CALCULATIONAL PROCEDURE
- FLAW SIZE CHARACTERIZATION
- STRESS INTENSITY EVALUATION
- CRACK GROWTH MODEL
- LEAK DETECTION
- FAILURE MODEL
- ASSUMPTIONS AND OBSERVATIONS

CRACK GROWTH MODEL

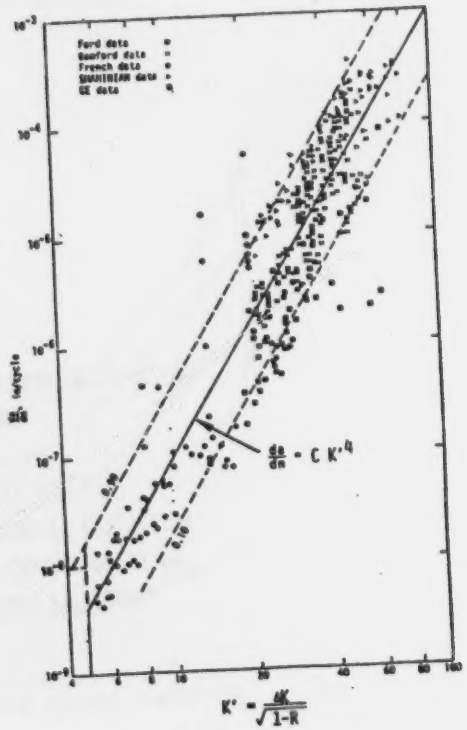
- PARIS EQUATION FOR 2-D CRACKS

  CRACK GROWTH RATE;

$$\frac{dA}{dN} = \begin{cases} 0 & K' < K_{TH} \\ C(K')^M & K' \geq K_{TH} \end{cases}$$

- M IS ASSUMED TO BE A FIXED VARIABLE
  - M=4 BASED ON THE AVAILABLE DATA
- C IS A RANDOM VARIABLE - BASED ON AVAILABLE DATA
  - DISTRIBUTION IS LOGNORMAL
    - MEDIAN = 9.14×10^{-12}
    - STD. DEV. = 3.02×10^{-11}
- K' IS A FUNCTION OF LOADING AND GEOMETRY
  - LOADING IS A RANDOM PROCESS
  - CRACK DIMENSIONS ARE RANDOM VARIABLES

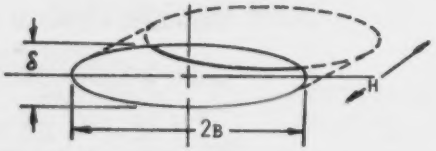
FATIGUE DATA ON 316 STAINLESS STEELS ARE LARGELY INDEPENDENT OF THE ENVIRONMENT OR WELD

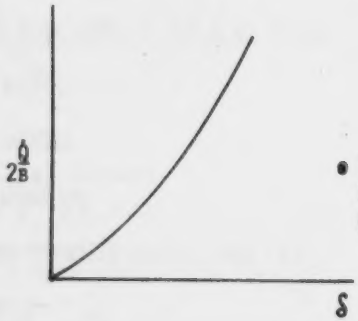


FRACTURE MECHANICS MODELING

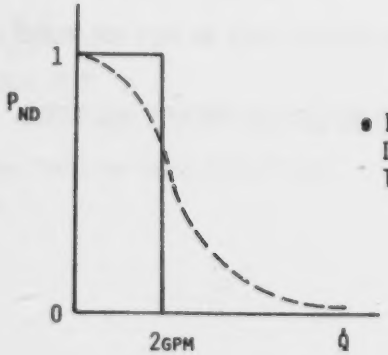
- CALCULATIONAL PROCEDURE
- FLAW SIZE CHARACTERIZATION
- STRESS INTENSITY EVALUATION
- CRACK GROWTH MODEL
- LEAK DETECTION
- FAILURE MODEL
- ASSUMPTIONS AND OBSERVATIONS

LEAK RATES ARE EVALUATED BY CALCULATING THE CRACK OPENING AREA USING LEFM CONCEPTS





- THE FLOW RATE IS A FUNCTION OF THE CRACK OPENING DISPLACEMENT AND THE CRACK LENGTH.



- IN THE CURRENT ASSESSMENT THE DETECTION PROBABILITY IS TAKEN TO BE 1 OR 0

FRACTURE MECHANICS MODELING

- CALCULATIONAL PROCEDURE
- FLAW SIZE CHARACTERIZATION
- STRESS INTENSITY EVALUATION
- CRACK GROWTH MODEL
- LEAK DETECTION
- FAILURE MODEL
- ASSUMPTIONS AND OBSERVATIONS

FAILURE MODEL CONSIDERS BOTH LEAK AND PIPE BREAK

- LEAKS GREATER THAN 2 GPM ARE ALWAYS DETECTED AND REPAIRED
- AFTER REPAIR, CRACK SIZE DISTRIBUTION IS RESET TO THE INITIAL SIZE DISTRIBUTION
- ONLY LOAD CONTROLLED STRESSES CAUSE LOCA
  - DEADWEIGHT
  - PRESSURE
  - SEISMIC
- UNSTABLE CRACK GROWTH IS DUE TO NET SECTION PLASTIC INSTABILITY
  - APPROXIMATIONS OF J AND T_{APPL} YIELD LARGER CRITICAL CRACK SIZE
- FLOW STRESS IS GIVEN BY $(\sigma_Y + \sigma_U)/2$
- FLOW STRESS IS NORMALLY DISTRIBUTED

- FAILURE MODEL COMPARISON:

PLASTIC INSTABILITY -vs- ELASTIC PLASTIC FRACTURE

- PIPE MATERIAL IS 316 STAINLESS STEEL
  - ASSUME WELDMENT PROPERTY IS EQUIVALENT TO BASE MATERIAL
- CONSIDER CRACK GEOMETRY WHICH WILL LEAD TO A LOCA WITHOUT PRIOR LEAKING
  - COMPLETE (OR NEARLY COMPLETE) CIRCUMFRENTIAL CRACK

(LEAKS DUE TO THROUGH WALL CRACKS ARE ASSUMED DETECTED AND REPAIRED)

1) NET-SECTION PLASTIC INSTABILITY

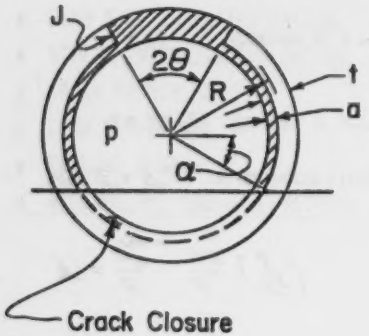
$$\sigma_{CRACKED\ SECTION} \geq \sigma_{FLOW} = 45\ KSI$$

2) ELASTIC - PLASTIC FRACTURE

$$J_{APPL} \geq J_{IC} = 3000\ IN\text{-}LB/IN^2$$

AND $T_{APPL} > T_{MAT'L} \sim 500$

PLASTIC LIMIT MOMEMT ANALYSIS*



GEOMETRY EMPLOYED IN TEARING ANALYSIS

$\bar{a} = \frac{a}{t}$

$\bar{p} = \frac{\sigma}{\sigma_0}$

$\theta = 0$

$J = \sigma_0 R \; F_J(\theta, \bar{a}, \bar{p}) \cdot \phi$

F_J IS ON THE ORDER OF UNITY ($1 < F_F < 2$)

$\phi_{PLASTIC} = \frac{ML}{EI}$; @ LIMIT MOMENT $\phi = \frac{M_p L}{EI}$

$J \sim \frac{\sigma_0 R}{EI} M_p L$

$\sim 20000 \; \frac{in\text{-}lb}{in^2} > J_{IC}$

*REF. TADA, PARIS, AND GA[illegible], NUREG/CR-0838

PLASTIC LIMIT MOMENT ANALYSIS.....CONT.

$$T_{APPL} \sim \frac{1}{3} F_1(\theta, \bar{a}, \bar{p}) \frac{L}{R} + F_2(\theta, \bar{a}, \bar{p}) \frac{JE}{\sigma_0^2 R}$$

F_1 ON ORDER OF 1

F_2 ON ORDER OF 0 - .5 (ASSUME .2)

$$T_{APPL} \sim \frac{1}{3} \frac{L}{R} + .2 \frac{JE}{\sigma_0^2 R}$$

$$\sim 5 + 20 = 25 \ll T_{MAT'L}$$

THUS, ALTHOUGH CRACK GROWTH MAY INITIATE DUE TO "J" OVERLOAD AT PLASTIC INSTABILITY, CRACK EXTENSION DUE TO TEARING IS STABLE.

FURTHER COMPARISON OF FAILURE MODELS INCLUDE:

- LINEAR ELASTIC BOUND ON J AND T
  - USE ELASTIC STRESS INTENSITY SOLUTION, K_I, TO ESTIMATE J_K AND TEARING FOR CIRCUMFRENTIAL CRACK

$$J = \frac{K_I^2}{E}, \quad K_I = \sigma\sqrt{a}\, f\left(\frac{a}{t}\right)$$

$$T = \frac{E}{\sigma_{FLOW}^2} \frac{\partial}{\partial a}\left(\frac{K_I^2}{E}\right)$$

  - FOR $\sigma = \sigma_{FLOW}$ ON UNCRACKED LIGAMENT

    $J \leq 200 \frac{in\text{-}k}{in^2}$ (MAXIMUM AT $A \sim \frac{t}{2}$)

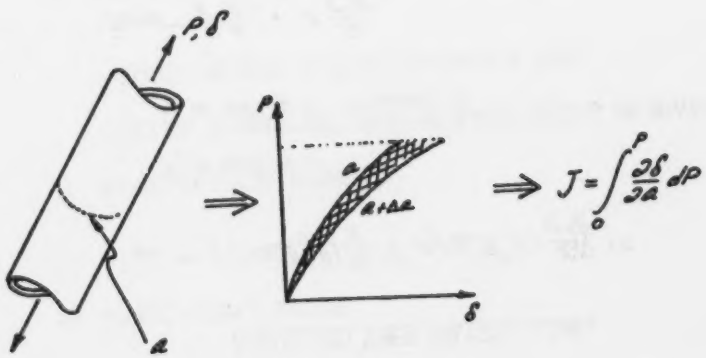
    $T < 10$ (MAXIMUM AT $A = 0$)

ELASTIC-PLASTIC ASSESSMENT OF FAILURE CRITERIA IS BASED ON POTENTIAL ENERGY DEFINITION OF J

- $$J = -\frac{\partial U}{\partial A}$$

WHERE U = POTENTIAL ENERGY IN CRACKED BODY

A = CRACK AREA



- USE ELASTIC PLASTIC COMPUTER CODE NIKE2D ON AXISYMMETRIC CIRCUMFRENTIALLY CRACKED PIPE

FRACTURE MECHANICS MODELING

- CALCULATIONAL PROCEDURE
- FLAW SIZE CHARACTERIZATION
- STRESS INTENSITY EVALUATION
- CRACK GROWTH MODEL
- LEAK DETECTION
- FAILURE MODEL
- ASSUMPTIONS AND OBSERVATIONS

IN MAKING THE DECOUPLING ASSESSMENT VARIOUS ASSUMPTIONS WERE MADE AND OBSERVATIONS NOTED.

- SIMPLIFY ANALYSIS
- ELIMINATE UNPRODUCTIVE EVALUATIONS
- BEST ESTIMATE IN LITERATURE
- REQUIRED DATA NOT AVAILABLE

ASSUMPTIONS AND OBSERVATIONS ARE ASSOCIATED WITH:

- CRACK CONFIGURATION AND EXISTANCE
- GROWTH MODEL
- STRESS ANALYSIS
- FAILURE MODELS

BASIC ASSUMPTION:

FAILURE IS DUE TO FATIGUE CRACK GROWTH OF CRACK LIKE DEFECTS WHICH ARE CONFINED TO THE GIRTH WELDED BUTT JOINTS.

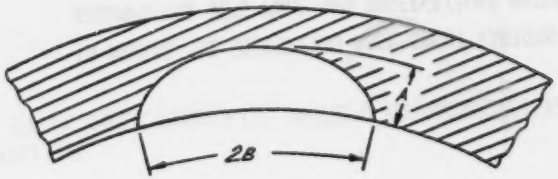
- CRACK SIZE DISTRIBUTION CAN BE ADEQUATELY CHARACTERIZED.
- NO DISTINCTION IS MADE BETWEEN THE SHOP OR FIELD WELDS.
- INITIAL CRACK SIZE DISTRIBUTION IS INDEPENDENT OF CRACK LOCATION.
- CRACKS DO NOT INTERACT - AT MOST THERE IS ONE CRACK PER JOINT.
- THE MAXIMUM MOMENT STRESS IS ASSUMED TO ACT ACROSS THE CRACK; I.E., MAXIMUM STRESS LOCATES CIRCUMFERENTIAL POSITION OF CRACK.
- CRACK AREA DOES NOT INFLUENCE LOCAL STRESSES
  - THIS MAY BE ACCOUNTED FOR BY ALTERING THE LOCAL MOMENT OF INERTIA WHILE ASSUMING THAT THE REMOTE APPLIED MOMENT IS UNAFFECTED.

$$\sigma_{CRACKED} = \frac{Mc}{I_{CRACKED}}$$

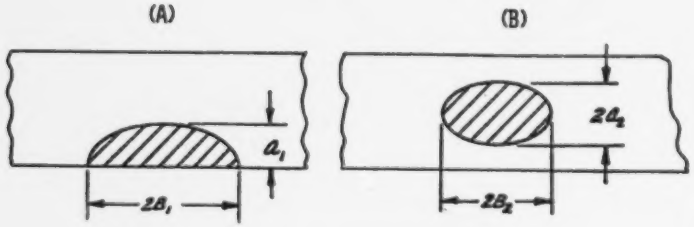
ASSUMPTIONS (CONT.)

ALL CRACKS ARE IN THE CIRCUMFERENTIAL ORIENTATION

- PROBABLE CRACK LOCATIONS WHICH ARE NOT CONSIDERED IN THE CURRENT ASSESSMENT INCLUDE:
  - LONGITUDINAL WELDS IN ELBOWS
  - NOZZLE CORNERS
  - BI-METALLIC TRANSITION JOINTS I.E., RVP TO SAFE END
  - BASE METAL
- CRACK SHAPE IS ELLIPTICAL
- ASPECT RATIO: $\frac{B}{A} \geq 1.$
- ONLY SURFACE CRACKS ARE EVALUATED
  - EQUIVALENT AREA SUBSURFACE CRACKS WILL RESULT IN A SUBSTANTIALLY REDUCED STRESS INTENSITY.



FOR EQUIVALENT AREA FLAWS, THE MAXIMUM STRESS INTENSITY FOR A SUBSURFACE FLAW IS SIGNIFICANTLY LESS THAN THAT OF A SURFACE FLAW.



- o For $2B_1 = 2B_2$ $A_1 = 2A_2$; ASSUME B >> A
- o ASSUME $\sigma < .3\sigma_y$
- o CONSIDER MEMBRANE STRESS ONLY; I.E., NO BENDING

$$K_I = M_M \sigma \sqrt{\frac{\pi a}{Q}}$$

THEN,	$\frac{A_1}{T} \sim 0$	$\frac{A_1}{T} = .35$
$\frac{K_{SURFACE}}{K_{EMBED}}$	1.414	2.54
$\left\lvert \frac{K_{SURFACE}}{K_{EMBED}} \right\rvert^4$	4	41.6

STRESSES RESULT FROM LOADING DUE TO PRESSURE, DEAD WEIGHT, OPERATING TRANSIENTS AND SEISMIC EVENTS

- STRESSES NOT EVALUATED IN THE CURRENT ASSESSMENT INCLUDE:
  - VIBRATION STRESS
  - RESIDUAL STRESS
  - TORSIONAL AND TRANSVERSE SHEAR (THESE STRESSES ARE SMALL COMPARED TO THE AXIAL STRESSES IN THE PRESENT STUDY)
  - WATER HAMMER
  - OTHERS ???
- DESIGN AND/OR CONSTRUCTION ERRORS ARE NOT ESTIMATED
- CONSERVATIVE DESIGN TEMPERATURE AND PRESSURE CHANGES ARE EMPLOYED IN THE STRESS ANALYSIS
- TRANSIENTS OCCUR AS A STATIONARY POISSON PROCESS REFLECTING ZION OPERATING HISTORY FOR 7 YEARS OF OPERATION
- EARTHQUAKES BELOW .06G HAVE NEGLIGIBLE EFFECT ON CRACK GROWTH
- FREE FIELD PEAK ACCELERATION .85G = SSE

SUBCRITICAL CRACK GROWTH IS DUE TO FATIGUE

STRESS CORROSION CRACKING AND CORROSION FATIGUE HAVE NOT BEEN OBSERVED IN PWR PRIMARY COOLANT PIPE

- PARIS MODEL ADEQUATELY DESCRIBES CRACK GROWTH
- FATIGUE BEHAVIOR FOR THE ELLIPTICAL SHAPE FLAW CAN BE PREDICTED BY LAB TESTS ON ONE DIMENSIONAL CRACKS
- CRACK RETARDATION DUE TO HIGH STRESS CYCLES IS NOT CONSIDERED
- ENHANCED CRACK GROWTH DUE TO LARGE $\Delta K'$ IS NOT CONSIDERED
- EFFECTS OF ENVIRONMENT AND/OR MATERIAL VARIATION ARE ACCOUNTED FOR IN DISTRIBUTION OF C
- NO VIBRATION STRESSES ARE CONSIDERED IN THE CURRENT ASSESSMENT, HOWEVER, THESE MAY BE IMPORTANT
- STEADY STATE THERMAL FLUCTUATIONS ARE FOUND TO RESULT IN ΔK BELOW THRESHOLD AND THUS DO NOT INFLUENCE CRACK GROWTH
- THROUGH WALL CRACK LENGTHS ARE SET EQUAL TO THE INSIDE WALL CRACK LENGTH

LAWRENCE LIVERMORE NATIONAL LABORATORY

LOAD COMBINATION PROGRAM

PROJECT 1
EVENT DECOUPLING

NUMERICAL SIMULATION

S. C. LU

8TH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING

GAITHERSBURG, MARYLAND

OCTOBER 31, 1980

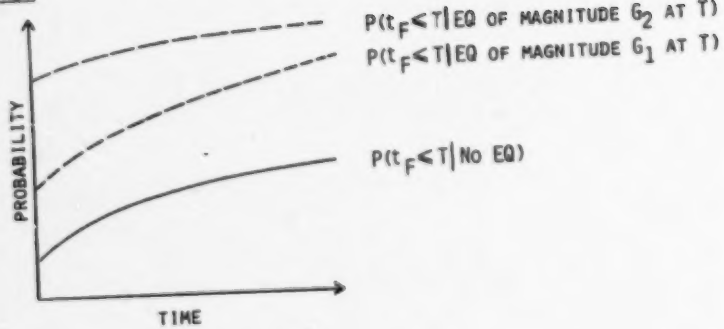
ASSUMES

(1) SINGLE WELD JOINT

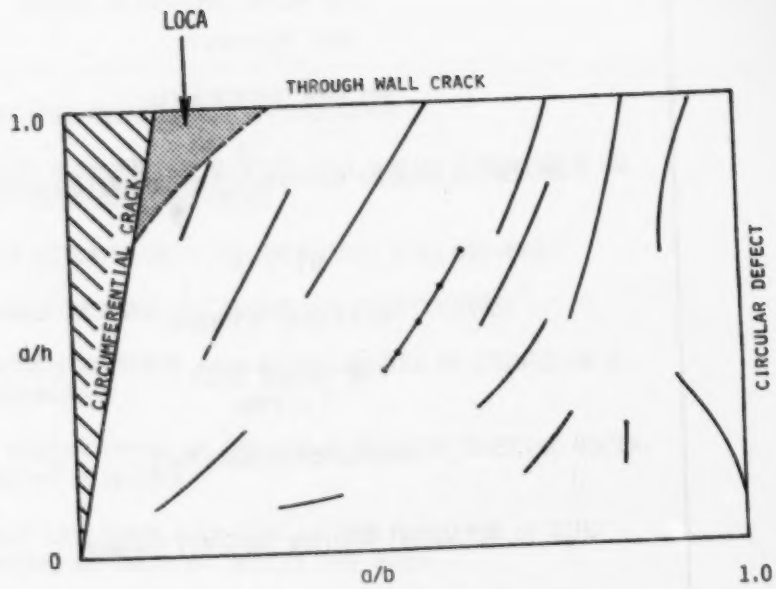
(2) INTERIOR SEMI-ELLIPTICAL CIRCUMFERENTIAL SURFACE DEFECT

(3) FATIGUE CRACK GROWTH

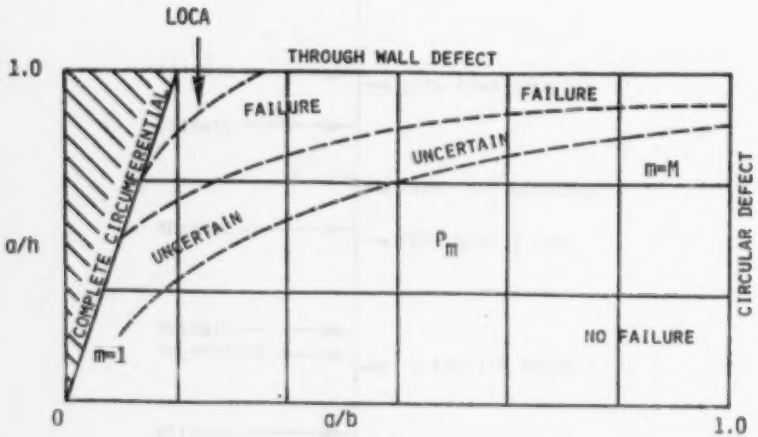
OUTPUT



- USER-ORIENTED
- COMPUTATIONALLY-EFFICIENT



$$P(t_F \leq T) \sim \frac{N_F(T)}{N}$$

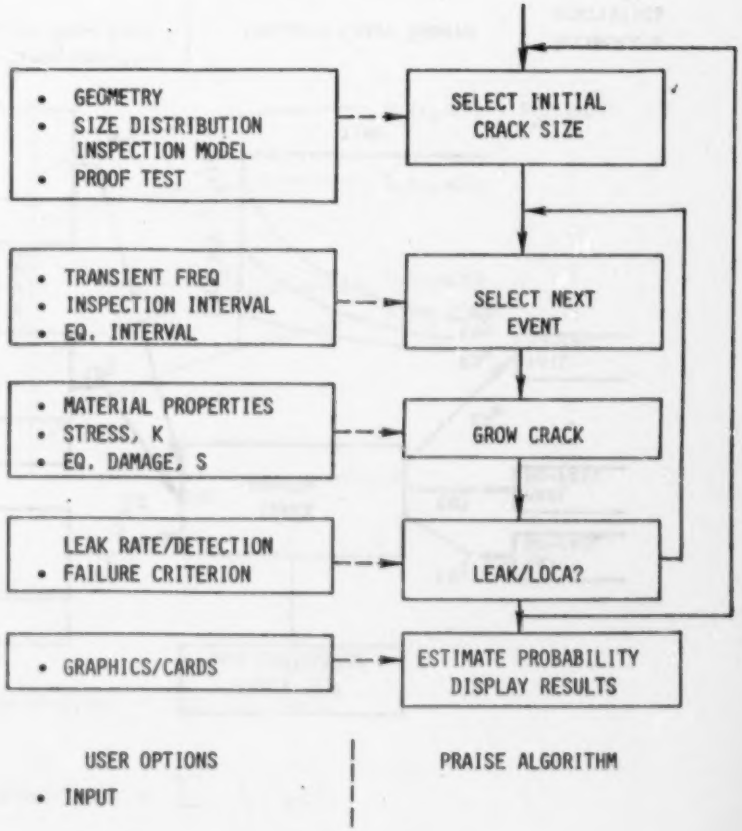


P_m = PROBABILITY OF INITIAL CRACK LYING IN CELL m

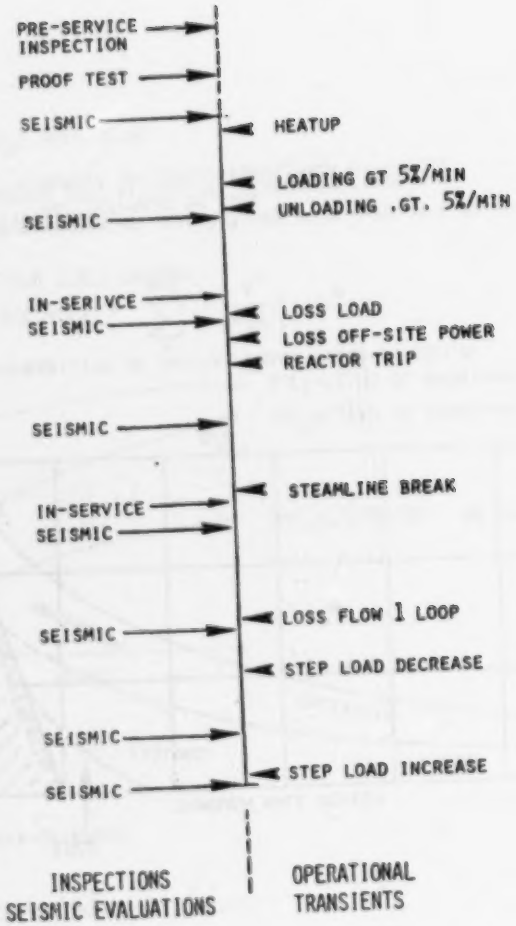
$$P(t_F < t) = \sum_{m=1}^{M} \frac{N_F(m,t)}{N_m} P_m$$

$N_F(m,t)$ = NUMBER OF REPLICATIONS FROM m-TH CELL IN WHICH FAILURE HAS OCCURRED BY TIME

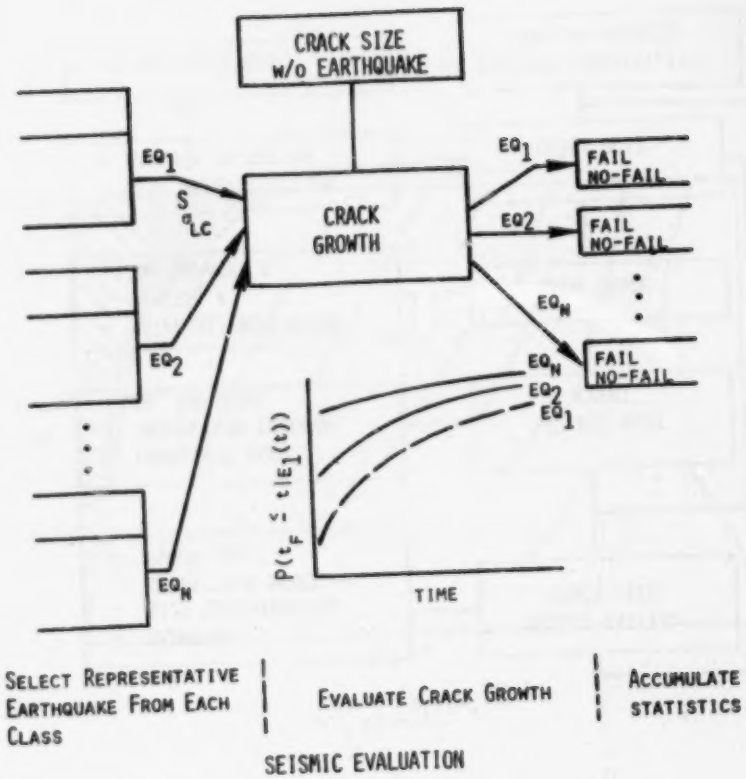
N_m = NUMBER OF REPLICATIONS FROM m-TH CELL



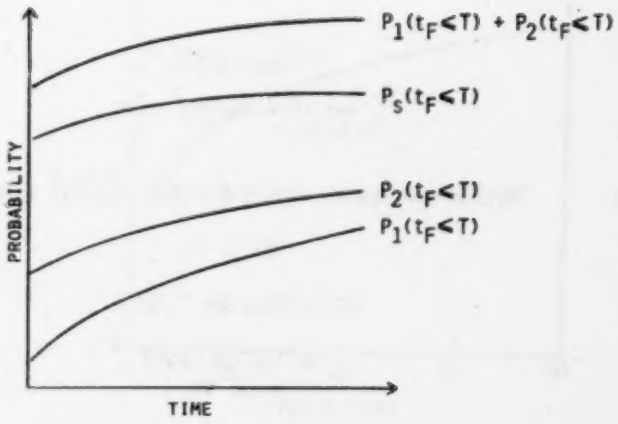
SIMPLIFIED PRAISE FLOWCHART



TRANISENT EVENT SEQUENCE IN PRAISE



SEISMIC EVALUATION



IN GENERAL,

$$\text{LOWER BOUND} = \max_j \; P_j(t_F < T)$$

$$\text{UPPER BOUND} = \sum_j P_j(t_F < T)$$

95-76

LAWRENCE LIVERMORE NATIONAL LABORATORY

LOAD COMBINATION PROGRAM

PROJECT 1
EVENT DECOUPLING

RESULTS AND CONCLUSION

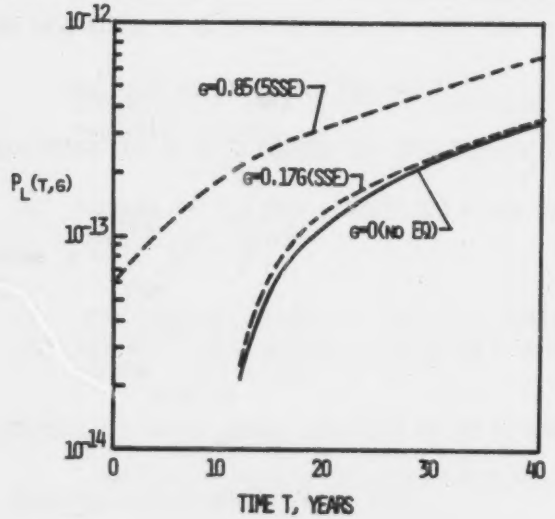
S. C. LU

8TH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING

GAITHERSBURG, MARYLAND

OCTOBER 31, 1980

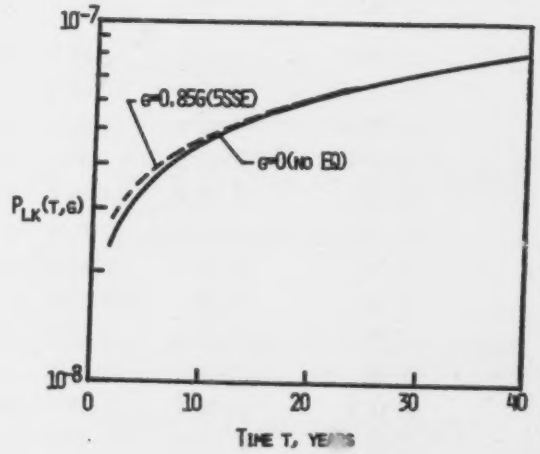
PROBABILITY OF A DIRECT LOCA WITH OR WITHOUT EARTHQUAKES (JOINT 1)



PROBABILITY OF A DIRECT LOCA AT OR BEFORE TIME τ INCLUDING THE EFFECT OF AN EARTHQUAKE OF MAGNITUDE G AT TIME $\tau = P_L(\tau,G)$

PROBABILITY OF A LOCA IN THE PLANT LIFE GIVEN NO EARTHQUAKE $= P_{L/\bar{E}} = P_L(40,0)$

PROBABILITY OF A LEAK WITH OR WITHOUT EARTHQUAKES (JOINT 1)



PROBABILITY OF A LEAK AT OR BEFORE TIME τ INCLUDING THE EFFECT OF AN EARTHQUAKE OF MAGNITUDE G AT TIME $\tau = P_{LK}(\tau,G)$

PROBABILITY OF A LEAK IN THE PLANT LIFE GIVEN NO EARTHQUAKE $= P_{LK/\bar{E}} = P_{LK}(40, 0)$

EARTHQUAKES HAVE VERY LITTLE ON LEAK PROBABILITY

PROBABILITY OF AN EARTHQUAKE INDUCED LOCA IN PLANT LIFE

1. THE PROBABILITY OF A LOCA AT TIME τ DUE TO AN EQ OF MAG G AT TIME τ

$$\bar{P}_{L/E}(\tau,G) = P_L(\tau,G) - P_L(\tau,0)$$

2. PROBABILITY OF AN EQ OF MAG G OR GREATER IN A YEAR, $P_E(G)$ - OBTAINED FROM THE SEISMIC HAZARD CURVE

3. PROBABILITY OF AN EQ INDUCED LOCA GIVEN ONE EQ IN PLANT LIFE

$$P_{L/E} = 1/40 \int_{.07G}^{3.0G} \int_{0}^{40} \bar{P}_{L/E}(\tau,G)\, P_E'(G)/K \, d\tau \, dG$$

WHERE $K = \int_{.07G}^{3.0G} P_E'(G)\, dG$

4. ABS PROBABILITY OF AN EQ INDUCED LOCA IN PLANT LIFE (40 YRS)

$$.7 * P_E(.07G) * 40 \; \mathrm{EXP}(-.7 * P_E(.07G) * 40) \; P_{L/E}$$

5. ABS PROBABILITY OF AN LOCA AND NO EQ IN PLANT LIFE

$$\mathrm{EXP}(-.7 * P_E(.07G) * 40) \; P_{L/\bar{E}}$$

NOTE : $P_E(.07G) = 10^{-2}$

RESULTS INDICATE LOCA PROBABILITIES ARE SMALL

JOINT	ABS $P_{L/\bar{E}}$	ABS $P_{L/E}$	ABS $(P_{L/\bar{E}} + P_{L/E})$
1	2.68-13	1.08-15	2.69-13
2	4.49-15	3.70-18	4.49-15
3	1.15-15	8.50-18	1.16-15
4	4.69-15	1.40-16	4.83-15
5	1.19-15	5.55-17	1.24-15
6	5.89-15	1.17-17	5.96-15
7	4.96-15	3.98-18	4.96-15
8	1.35-15	2.90-19	1.35-15
9	3.11-15	4.70-17	3.16-15
10	2.01-15	4.00-17	2.05-15
11	0	5.78-17	5.78-17
12	2.43-15	6.08-15	8.51-15
13	1.84-15	4.09-15	5.93-15
14	1.82-16	2.71-17	2.09-16
4 Σ	1.20-12	4.66-14	1.25-12

(ABS $P_{L/E}$) / ABS $(P_{L/\bar{E}} + P_{L/E})$ = 4%

SUMMARY: LOCAs DIRECTLY INDUCED BY EARTHQUAKE

- o PROBABILITY OF A LOCA EITHER WITH OR WITHOUT EARTHQUAKE IS SMALL.
- o PROBABILITY OF A LEAK IS MORE LIKELY THAN THAT OF A COMPLETE PIPE RUPTURE.
- o LEAK DETECTION PROBABILITY, PRESERVICE INSPECTIONS, AND PROOF TESTING ARE IMPORTANT FACTORS IN REDUCING RROBABILITY OF A LOCA
- o SEISMIC STRESSES ARE NOT LARGE AND DO NOT SIGNIFICANTLY INCREASE THE LOCA PROBABILITY.
- o HEAT-UP TO COOL-DOWN CYCLE DOMINATES THE STRESS HISTORY.

SUMMARY: LOCAs INDIRECTLY INDUCED BY EARTHQUAKES

- o MOST OF THE LOCAs ARE NOT DOUBLE-ENDED RCL PIPE BREAKS
- o TRANSIENT INITIATED EVENTS
  - FIRES, ELECTRICAL FAILURES, ETC.
  - SEQUENCES INVOLVING PORV'S INABILITY TO RECLOSE OR BE BLOCKED ARE THE DOMINANT SOURCES.
  - PROBABILITY 10^{-6} (GIVEN AN SSE)
- o MISSILES
  - FAILURES OR RUPTURES OF OVERHEAD CRANE CONTAINMENT DOME, RCP, RCL STOP VALVES, ETC.
  - LESS LIKELY THAN TRANSIENT EVENTS.
  - PROBABILITY 10^{-9} (GIVEN AN SSE)
- o EXPLOSIONS
  - HYDROGEN, NITROGEN, BORIC ACID, ETC.
  - THE LEAST LIKELY EVENTS.

DECOUPLING LARGE LOCA AND SSE LOADS APPEARS TO BE APPROPRIATE FOR ZION-1 NUCLEAR POWER STATION

WHY ?

- o PROBABILITY OF SIMULTANEOUS LARGE LOCA AND EARTHQUAKE IS SMALL (SUBJECT TO REVIEW ON RESULTS AND ASSUMPTIONS USED IN THIS EVALUATION)
- o ESTIMATES ARE BASED ON REALISTICAL INPUTS, REASONABLE ASSUMPTIONS, AND ADVANCED METHODOLOGY.

BUT.....

- o REQUIRE FURTHER IMPROVEMENTS TO ESTABLISH BETTER CONFIDENCE
- o RELIABILITY OF STRUCTURES AND SYSTEMS WHICH MAY CAUSE INDIRECTLY INDUCED LOCAs DURING EARTHQUAKES SHOULD BE PROPERLY MAINTAINED.
- o A STUDY IS REQUIRED TO ANSWER A HYPOTHETICAL QUESTION: WHAT WOULD BE THE CONSEQUENCIES IF ZION WERE TO BE DESIGNED FOR DECOUPLING.

MORE PROBLEMS OF EVENT DECOUPLING NEED TO BE SOLVED

- o IMPROVEMENT OF THE CURRENT DECOUPLING METHODOLOGY
- o EARTHQUAKE INDUCED SMALL LOCA ASSOCIATED WITH PIPE BREAK IN BRANCH LINES OF A PWR REACTOR COOLING LOOP
- o LOCA - EARTHQUAKE DECOUPLING OF A BWR PLANT
- o EXTENDED CONCLUSIONS OF LOCA-EARTHQUAKE DECOUPLING TO OTHER PWR AND BWR PLANTS
- o OTHER LOW PROBABILITY EVENT COMBINATIONS

LAWRENCE LIVERMORE NATIONAL LABORATORY

LOAD COMBINATION RESEARCH PROGRAM

PROBABILISTIC METHODOLOGY FOR EVALUATION OF COMPONENT DESIGNS

PRESENTED BY

C. A. CORNELL, M.I.T.

PRESENTATION TO

8TH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING

GAITHERSBURG, MARYLAND

OCTOBER 31, 1980

PROBABILISTIC METHODOLOGY FOR EVALUATION OF NUCLEAR COMPONENT DESIGNS

C. Allin Cornell

Massachusetts Institute of Technology

The load combination methodology development project at Lawrence Livermore National Laboratory has two objectives:

- To provide a procedure to evaluate the reliability implied by a set of design criteria. This will be useful to the NRC in its licensing review.
- To derive load and resistance factor values in load combinations appropriate for the design of nuclear components to a target limit state probability.

In order to achieve these objectives, we first need to estimate the limit state probability, P_F, of a component. The method of evaluating P_F is based on the probability distribution of resistance R of the component and of extreme combined load effect (response) on the component. The probability distribution of the load effect is developed from convolving probability distribution of static load effects and the extreme of the combined time-varying "static" and/or dynamic load effects. The latter distribution is approximated by a method which makes use of the mean up-crossing rate $\nu(y)$ as a function of the response y, and the a.p.t distribution of response, as described by the probability density function $f_Y(y)$. For the purposes of calculating the probability distribution of the combined response three types of loads are identified: static, continuous and intermittent.

The method of evaluating the probabilities of reaching different limit states is illustrated using the example of an Essential Service Water line in a pressurized water reactor.

OUR OBJECTIVE IS TO DEVELOP METHODOLOGIES FOR, EVALUATING AND DESIGNING FOR MULTIPLE COMBINED LOADINGS

MORE SPECIFICALLY WE SEEK TO DEVELOP METHODOLOGIES FOR

- EVALUATION:
  - A. TO EVALUATE THE RELIABILITY OF A GIVEN COMPONENT UNDER A FULL SPECTRA OF MULTIPLE STATIC AND DYNAMIC, CONTINUOUS AND INTERMITTENT LOADING HISTORIES.
  - B. TO EVALUATE THE RELIABILITY IMPLICATIONS OF A GIVEN DESIGN CRITERIA.
- CODE GENERATION

  TO SYSTEMATICALLY DETERMINE THE DESIGN CRITERIA THAT ACHIEVE A TARGET COMPONENT REALIABILITY.

THE METHODOLOGY DEVELOPMENT IS CONDUCTED IN THE FOLLOWING CONTEXT:

- o PROBABILISTIC EVALUATIONS
- o PROBABILISTIC TREATMENT OF:
  - STATIC AND DYNAMIC LOADS
  - CONTINUOUS AND INTERMITTENT LOADS
  - NORMAL PLUS UNUSUAL INTERNAL AND EXTERIOR LOADS
- o DESIGN CRITERIA WHICH ARE CONVENTIONAL (DETERMINISTIC) SETS OF CHECKING EQUATIONS, E.G.

$$S_H > P + W$$
$$1.25\ S_H > P + W + OBE$$

TYPICAL QUESTIONS THAT ARE ADDRESSED:

- o TO WHICH ASME SERVICE LEVEL SHOULD THE COMBINATION HTR + OBE BE ASSIGNED?
- o SHOULD THE DESIGN BE CHECKED AT ALL FOR THE COMBINATION HTR + OBE , OR SIMPLY EACH SEPARATELY?
- o WHAT VALUE OF THE FACTOR ON YIELD STRESS AND WHAT VALUES OF NOMINAL LOADS AND/OR LOAD FACTORS SHOULD BE USED TO GENERATE DESIGNS?

TODAYS'S PRESENTATION WILL CONSIST OF:

I CRITERIA EVALUATION METHODOLOGY

A. TECHNIQUES FOR EVALUATING THE RELIABILITY OF A GIVEN COMPONENT UNDER MULTIPLE STOCHASTIC LOADINGS. . . C.A. CORNELL

B. ANALYSIS TECHNIQUES FOR EVALUATING THE RELIABILITIES IMPLIED BY A GIVEN SET OF DESIGN CRITERIA WHEN APPLIED TO A FAMILY OF COMPONENTS

AND

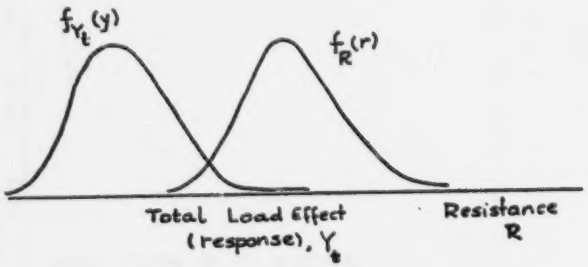
APPLICATIONS TO AN ESSENTIAL SERVICE WATER LINE. . . . . M. K. RAVINDRA

II GENERATION OF OPTIMAL LOAD COMBINATION CRITERIA.

A. TECHNIQUES FOR PRODUCING THE SET OF DESIGN CRITERIA WHICH MEET GIVEN RELIABILITY OBJECTIVES - ILLUSTRATED WITH A SIMPLE EXAMPLE. . . . M.W. SCHWARTZ

B. APPLICATION TO AN ESSENTIAL SERVICE WATER LINE . . . . . M.K RAVINDRA

LIMIT STATE PROBABILITY



$$P_{f_k} = P[Y_t > R]$$

$$= \int_0^\infty [1-F_{Y_t}(r)]\ f_R(r)dr$$

RESPONSE

$$Y_t = Y_s + Y_c + Y_{int} = Y_s + Y$$

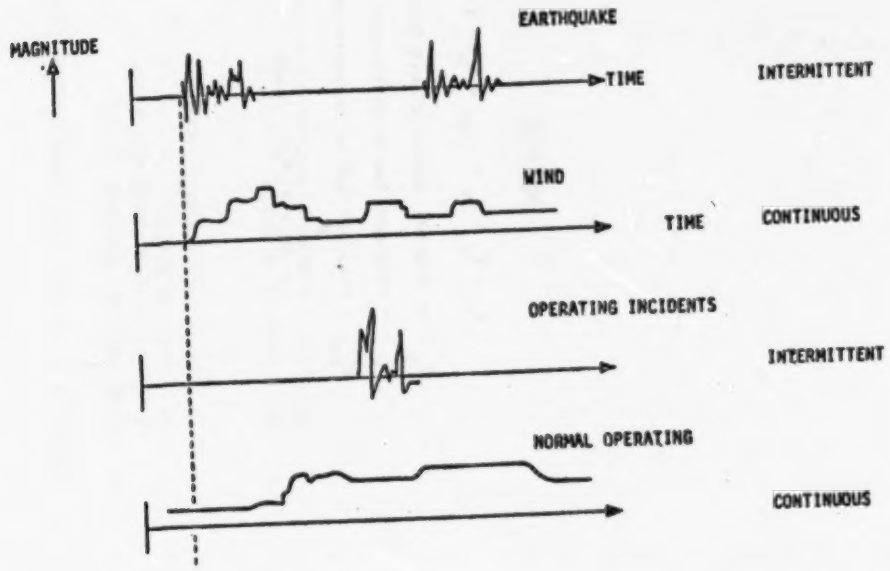
Y_s = response due to static loads

Y_c = response due to continuous loads

Y_{int} = response due to intermittent loads

Y = combined response due to continuous and intermittent loads

EXAMPLES OF INITIAL LOADS

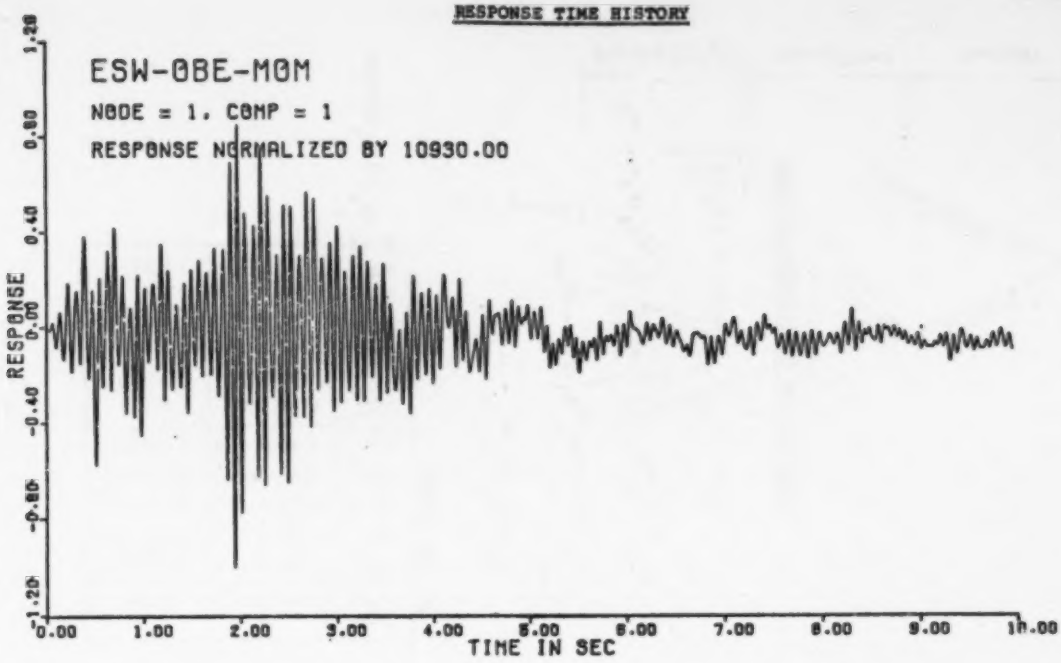


RESPONSE DUE TO STATIC LOADS

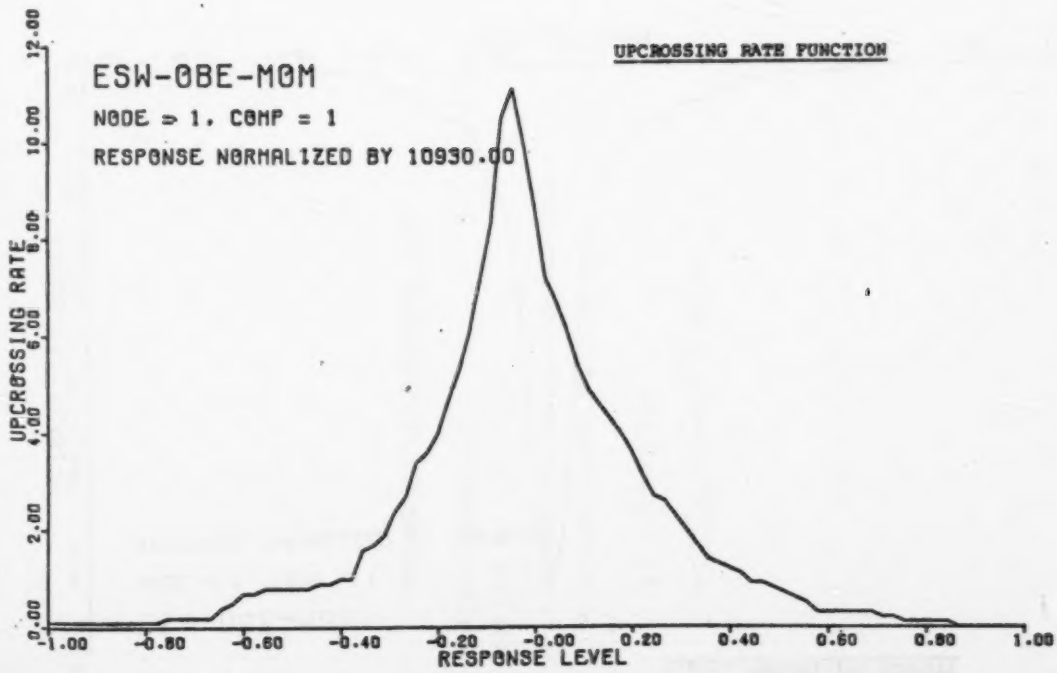
$$X_s = c_1 X_1 + c_2 X_2$$

$$f_{X_s}(y) = \frac{1}{c_1 c_2} \int f_{X_1}\left(\frac{y-u}{c_1}\right) f_{X_2}\left(\frac{u}{c_2}\right) du$$

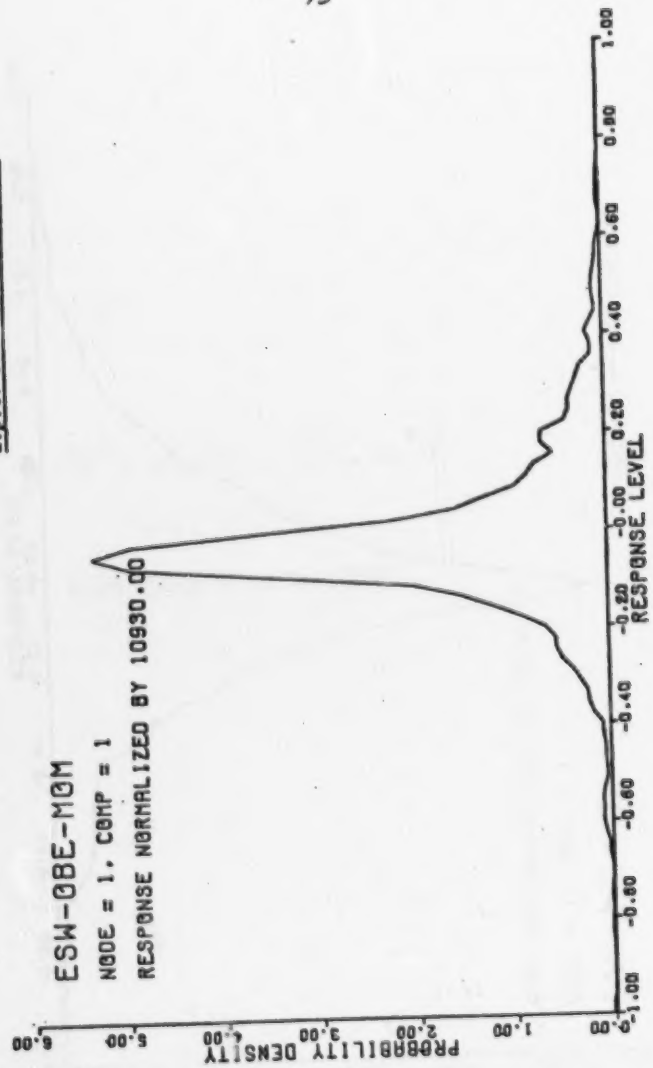
RESPONSE TIME HISTORY



UPCROSSING RATE FUNCTION



a.p.t. PROBABILITY DENSITY



RESPONSE DUE TO CONTINUOUS LOADS

$$Y_c(t) = \sum_{\text{all } i} c_i X_i(t) = \Sigma Y_i(t)$$

$$\nu_{Y_c} = \nu_{Y_1} * f_{Y_2+Y_3+\ldots+Y_n} + \nu_{Y_2} * f_{Y_1+Y_3+\ldots+Y_n}$$

$$+\ldots+\nu_{Y_n} * f_{Y_1+Y_2+\ldots+Y_{n-1}}$$

$$f_{Y_c} = f_{Y_1} * f_{Y_2} * f_{Y_3} * \ldots * f_{Y_n}$$

For two loads:

$$\nu_{Y_c}(y) = \int_{-\infty}^{\infty} \nu_{Y_1}(u)\, f_{Y_2}(y-u)\,du + \int_{-\infty}^{\infty} \nu_{Y_2}(u)\, f_{Y_1}(y-u)\,du$$

RESPONSE DUE TO INTERMITTENT LOADS

Additional Parameters

Load Case q; $\nu_q(x)$, $f_q(x)$

λ_i = mean rate of occurrence of load event i

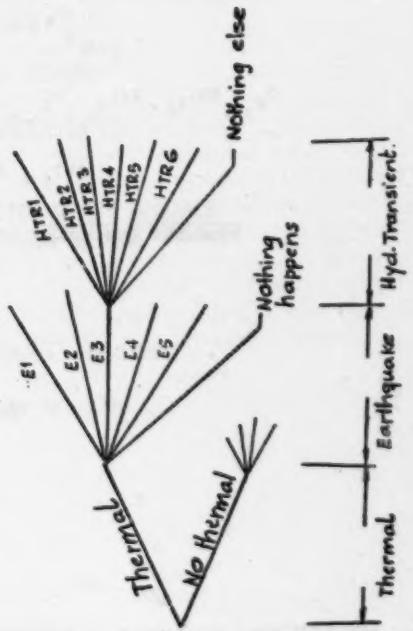
T_i = duration of load event i ($= \mu_i$)

$$\lambda_{1+2} = \lambda_1 \lambda_2 (\mu_1 + \mu_2)$$

$$\lambda_{1+2+\ldots+n} = (n-1)! \left\{ \prod_{i=1}^{n} (\lambda_i \mu_i) \right\} \sum_{i=1}^{n} \frac{1}{\mu_i}$$

LOAD EVENT TREE FOR INTERMITTENT LOADS FOR ESW PIPING EXAMPLE

Load Level	Magnitude	λ/yr
E1	0.10 g	8.00×10^{-4}
E2	0.20 g	1.10×10^{-4}
E3	0.30 g	0.55×10^{-4}
E4	0.40 g	0.30×10^{-4}
E5	0.50 g	0.05×10^{-4}
HTR1	0.25 sec	9.15
HTR2	0.50 sec	19.90
HTR3	1.00 sec	9.25
HTR4	1.50 sec	4.55
HTR5	2.00 sec	2.55
HTR6	2.50 sec	4.60



RESPONSES DUE TO INTERMITTENT LOADS

$$\nu_q(y) = p_1 p_2 [\nu_{Y_1} * f_{Y_2} + \nu_{Y_2} * f_{Y_1}] + p_1(1-p_2)\nu_{Y_1} + p_2(1-p_1)\nu_{Y_2}$$

$$f_q(y) = p_1 p_2 [f_{Y_1} * f_{Y_2}] + p_1(1-p_2) f_{Y_1} + p_2(1-p_1) f_{Y_2} + (1-p_1)(1-p_2)\,\delta(y)$$

$$p_i = \frac{\mu_i}{\mu_1 + \mu_2}$$

$$\nu_{Y_{int}}(y) = \sum_{q=1}^{Q} \lambda_q \nu_q(y) \mu_q$$

$$\mu_q = \Sigma \mu_i$$

$$f_{Y_{int}}(y) = \sum_{q=1}^{Q} f_q(y)\, \lambda_q \mu_q + \delta(y)\left[1 - \sum_{q=1}^{Q} \lambda_q \mu_q\right]$$

COMBINED RESPONSE

$$Y = Y_c + Y_{int}$$

$$\nu_Y(y) = \nu_{Y_c} * f_{Y_{int}} + \nu_{Y_{int}} * f_{Y_c}$$

$$F_Y(y) \simeq e^{-\nu_Y(y) T}$$

$$P_{f_k} = P[Y > R^*] = \int_0^\infty [1 - F_Y(r)]\, f_{R^*}(r)\, dr$$

$$P_F \approx \max_{k=1}^{K} P_{f_k}$$

LAWRENCE LIVERMORE NATIONAL LABORATORY

LOAD COMBINATION RESEARCH PROGRAM

FUNDAMENTALS OF LOAD COMBINATION METHODOLOGY

PRESENTED BY

M. W. SCHWARTZ

PRESENTATION TO

8TH WATER REACTOR SAFETY RESEARCH
INFORMATION MEETING

GAITHERSBURG, MARYLAND

OCTOBER 31, 1980

FUNDAMENTALS OF LOAD COMBINATION METHODOLOGY

SUMMARY

This presentation introduces the methodology for deriving optimized load combination formats for the design of nuclear power plant components. This process is the inverse of evaluating components probabilistically. It involves finding a set of load factors that will result in a design that meets a specified or target limit state probability. The methodology is outlined and illustrated with a simple example consisting of an array of pipes of varying diameter and length subjected simultaneously to dead weight, internal pressure and an initial velocity transient. Inherent in the methodology is an evaluation step which computes the limit state probability of a component whose configuration is defined. This evaluation is performed for a set of factored load combination equations. An optimized set of load factors is then derived based upon a scheme which minimizes the difference between the computed limit state probability and the target limit state probability for the entire array of pipe designs. This produces a set of load factors each of which corresponds to a particular level of reliability desired for the component. It is important to emphasize that this methodology is consistent with conventional design practice. The designer uses deterministic equations to check and generate designs. The probabilistically derived load factors are supplied by a separate code writing group.

LOAD COMBINATION METHODOLOGY DEVELOPMENT HAS THE FOLLOWING OBJECTIVES

- TO EVALUATE THE LIMIT STATE PROBABILITY OF A GIVEN COMPONENT SUBJECTED TO MULTIPLE LOADS.
- TO EVALUATE THE LIMIT STATE PROBABILITY IMPLIED BY A GIVEN DESIGN CRITERIA WITH EITHER UNIT OR ARBITRARY LOAD FACTORS.
- TO DETERMINE THE DESIGN CRITERIA THAT ACHIEVE SPECIFIED OR TARGET LIMIT STATE PROBABILITIES.

$$\phi_s M \geq \gamma_1 \gamma_1 + \gamma_2 \gamma_2 + \text{----}$$

DIFFERENT ASSUMPTIONS OF LOAD COMBINATIONS AND LOAD MAGNITUDES LEAD TO DIFFERENT LEVELS OF LIMIT STATE PROBABILITY

LAWRENCE LIVERMORE LABORATORY
University of California

IF WE FIX THE LIMIT STATE PROBABILITY AT SOME ACCEPTABLY LOW LEVEL WE CAN, FOR A SPECIFIED STRESS INTENSITY LIMIT, ESTABLISH A DESIGN CONFIGURATION WHOSE RESPONSE LEVELS CORRESPOND TO THE SPECIFIED OR TARGET LIMIT STATE PROBABILITY.

WE DO THIS BY ASSOCIATING AN APPROPRIATE FACTOR WITH EACH RESPONSE IN COMBINATION.

THUS THE DESIGN FORMAT EQUATION TAKES THE FORM

$$\phi S_M > \gamma_1 Y_1 + \gamma_2 Y_2 + \gamma_3 Y_3 + \ldots$$

LAWRENCE LIVERMORE LABORATORY
University of California

WE NOW ILLUSTRATE THE DERIVATION OF LOAD FACTORS WITH A SIMPLE EXAMPLE

- COMPONENT: SIMPLY SUPPORTED LENGTH OF PIPE AS SHOWN IN THE FOLLOWING FIGURE.

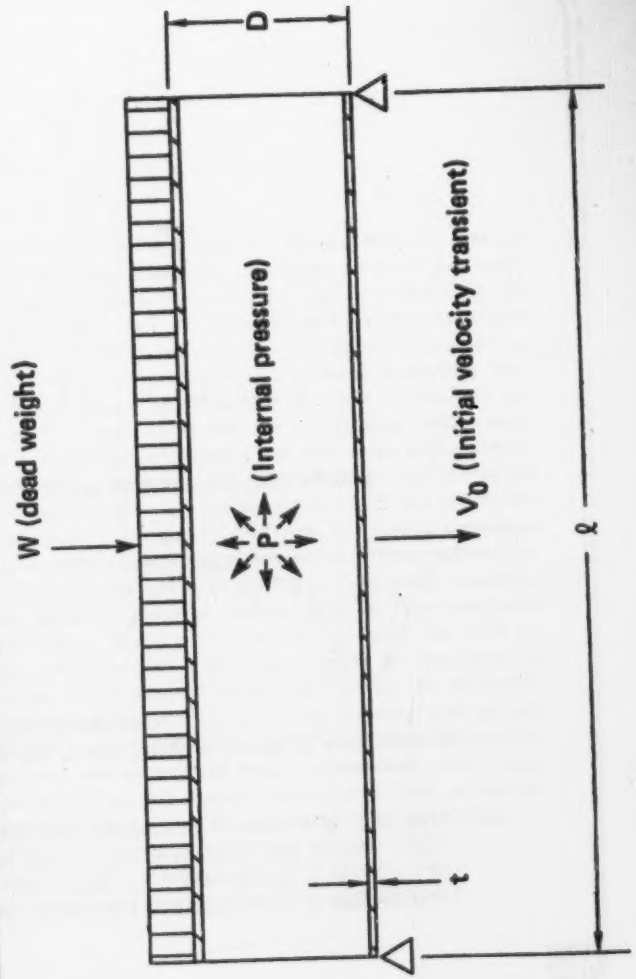
- LOADS: DEAD WEIGHT
  INTERNAL PRESSURE
  INITIAL VELOCITY TRANSIENT

- ASSUMPTIONS:

  ONLY LOADS TREATED AS RANDOM VARIABLES

  STATISTICAL INDEPENDENCE
  NORMAL DISTRIBUTIONS
  SMALL STANDARD DEVIATIONS

LOAD COMBINATION CONFIGURATION FOR SIMPLY SUPPORTED PIPE EXAMPLE



STEPS IN THE PROCEDURE FOR GENERATING LOAD COMBINATION FACTORS

1. Select a design format

$$\phi S_m = \gamma_1 c_1 V_0 + \gamma_2 c_2 \ell_2 + \gamma_3 c_3 p_3$$

2. Obtain a set of influence coefficients

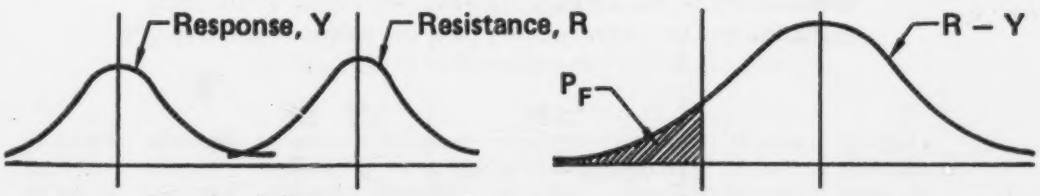
Load, L → [Influence coefficient c] → Response, Y = cL

3. Assume initial values for load and resistance factors $\phi, \gamma_1, \gamma_2, \gamma_3$

4. Adjust design parameters until the load combination equation is satisfied for the assumed values or load and resistance factors

STEPS IN THE PROCEDURE FOR GENERATING LOAD COMBINATION FACTORS (CONT'D)

5. Analyze the design probabilistically to determine its limit state probability, P_F



6. Repeat steps 4 and 5 for other components to obtain a set of limit state probabilities P_F corresponding to the assumed load and resistance factors

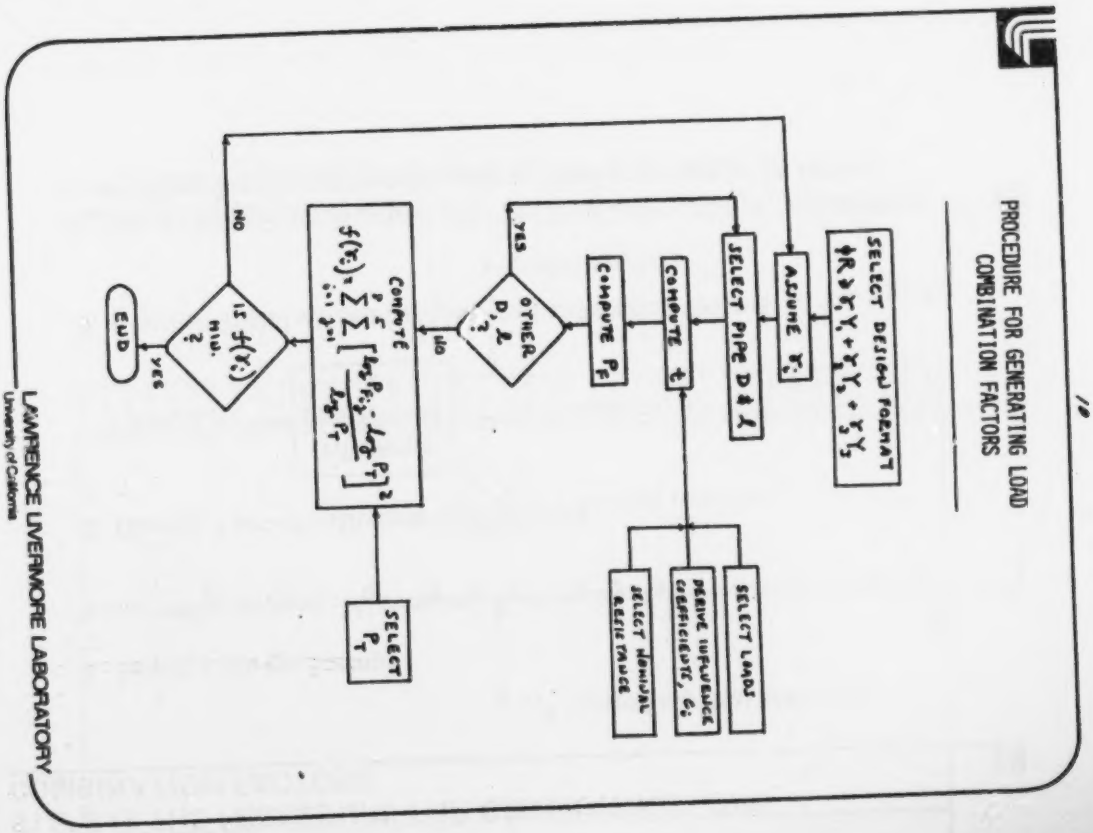
STEPS IN THE PROCEDURE FOR GENERATING LOAD COMBINATION FACTORS (CONT'D)

7. Select a target limit state probability, P_T, and set up the objective function

$$f(\gamma) = \sum_{i=1}^{8} \sum_{j=1}^{5} \left[\frac{\log P_{Fij} - \log P_T}{\log P_T} \right]^2$$

8. Select other values for load factors such that the objective function is minimized over the entire array of components

The load and resistance factors which minimize the objective function constitute an optimal set of design rules



Limit state probabilities corresponding to $\gamma_1 = \gamma_2 = \gamma_3 = 1$

Length (in.)	Pipe diameter (in.)							
	10.	12.	14.	16.	18.	20.	22.	24.
120.	5.6994E-03	4.9738E-03	6.1614E-03	5.0286E-03	5.2147E-03	8.8937E-03	7.0195E-03	5.1023E-03
240.	2.3799E-03	2.6540E-03	3.3247E-03	3.1919E-03	3.8129E-03	5.1931E-03	4.1396E-03	4.0209E-03
360.	1.0774E-03	1.2169E-03	1.4767E-03	2.0053E-03	2.2644E-03	2.5292E-03	2.5873E-03	2.7178E-03
480.	2.8326E-04	4.3484E-04	6.3211E-04	8.1391E-04	9.6302E-04	1.1098E-03	1.3694E-03	1.5159E-03
600.	8.2522E-05	1.4364E-04	2.1297E-04	3.0179E-04	4.0161E-04	5.0133E-04	6.2701E-04	7.6009E-04

Pipe Thickness (in.) corresponding to $\gamma_1 = \gamma_2 = \gamma_3 = 1$

Length (in.)	Pipe diameter (in.)							
	10.	12.	14.	16.	18.	20.	22.	24.
120.	0.355	0.447	0.487	0.597	0.664	0.624	0.737	0.895
240.	0.456	0.534	0.576	0.677	0.715	0.719	0.857	0.948
360.	0.533	0.653	0.725	0.739	0.809	0.872	0.963	1.042
480.	0.835	0.878	0.883	0.946	1.034	1.116	1.133	1.209
600.	1.870	1.407	1.273	1.290	1.324	1.396	1.400	1.440

Table 1.3.

LOAD FACTORS CORRESPONDING TO 1.0E-02 TARGET LIMIT STATE

GAMMA1 = 0.8747 GAMMA2 = 0.8733 GAMMA3 = 1.0941

LIMIT STATE PROBABILITIES CORRESPONDING TO 1.0E-02 TARGET LIMIT STATE

LENGTH(IN)	PIPE DIAMETER(IN)							
	10.	12.	14.	16.	18.	20.	22.	24.
120.	1.0184E-02	1.0496E-02	1.0219E-02	1.0966E-02	1.0080E-02	1.0032E-02	1.0075E-02	1.0788E-02
240.	1.0274E-02	1.0530E-02	1.0095E-02	1.0174E-02	1.0164E-02	0.9963E-03	1.0038E-02	1.0413E-02
360.	0.9935E-03	0.9968E-03	1.0148E-02	0.9788E-03	0.9888E-03	0.8187E-03	0.8871E-03	0.8115E-03
480.	0.8840E-03	0.8881E-03	0.8718E-03	0.9030E-03	0.7481E-03	0.8032E-03	0.8408E-03	0.8740E-03
600.	1.0218E-02	1.1018E-02	0.7128E-03	0.8287E-03	0.8779E-03	0.8174E-03	0.8788E-03	0.8105E-03

MEAN = 1.0040E-02 STD.DEV = 0.0049E-02 C.O.V. = 0.0000E-02

PIPE THICKNESS CORRESPONDING TO 1.0E-02 TARGET LIMIT STATE

LENGTH(IN)	PIPE DIAMETER(IN)							
	10.	12.	14.	16.	18.	20.	22.	24.
120.	3.0554E-01	4.2311E-01	4.7552E-01	5.7479E-01	6.4281E-01	6.0032E-01	7.0438E-01	8.6498E-01
240.	4.0008E-01	5.0175E-01	5.5738E-01	6.4844E-01	6.8411E-01	6.8888E-01	8.2282E-01	9.0746E-01
360.	4.8490E-01	6.1818E-01	6.8276E-01	7.0637E-01	7.7437E-01	8.3160E-01	9.0070E-01	9.8786E-01
480.	8.0108E-01	7.4883E-01	7.7742E-01	8.1377E-01	8.8838E-01	1.0032E+00	1.0832E+00	1.1502E+00
600.	1.7739E+00	1.0884E+00	1.0328E+00	1.0704E+00	1.1278E+00	1.2088E+00	1.2540E+00	1.2888E+00

LOAD FACTORS FOR THE CASE $\phi S_m = \gamma_1 Y_1 + \gamma_2 Y_2 + \gamma_3 Y_3$

P_T	γ_1	γ_2	γ_3	$\bar{P}_F$	$\sigma(P_F)$	COV	3σ NEP
10^{-2}	0.875	0.873	1.034	1.003×10^{-2}	6.000×10^{-4}	0.060	1.183×10^{-2}
10^{-3}	0.917	0.913	1.122	1.007×10^{-3}	1.076×10^{-4}	0.107	1.330×10^{-3}
10^{-4}	0.957	0.951	1.197	1.014×10^{-4}	1.632×10^{-5}	0.161	1.504×10^{-4}
10^{-5}	0.995	0.987	1.263	1.023×10^{-5}	2.282×10^{-6}	0.223	1.708×10^{-5}
10^{-6}	1.033	1.023	1.324	1.037×10^{-6}	3.051×10^{-7}	0.294	1.952×10^{-6}
10^{-7}	1.070	1.059	1.381	1.054×10^{-7}	3.970×10^{-8}	0.377	2.245×10^{-7}
10^{-8}	1.108	1.076	1.434	1.076×10^{-8}	5.099×10^{-9}	0.474	2.606×10^{-8}
——	1.0	1.0	1.0	2.571×10^{-3}	2.208×10^{-3}	0.859	——

13

USE OF FACTORED LOAD COMBINATION FORMAT IN DESIGN

- TARGET LIMIT STATE PROBABILITY = 10^{-6}
- DESIGN FORMAT:

$$1.5\, S_M = 1.05\, \sigma_1 V_0 + 1.02\, \sigma_2 \ell + 1.32\, \sigma_3 P$$

- DESIGN DATA:

V_0 = 13 IN/SEC

ℓ = 300 IN

P = 250 PSI

S_M = 16,500 PSI

D = 15 IN.

- RESULTS:

T = 0.776 IN.

LIMIT STATE PROBABILITY = 4.77×10^{-6}

14

1. We start with the selection of a design format

$$\phi S_m \geq \gamma_1 c_1 V_o + \gamma_2 c_2 \ell + \gamma_3 c_3 p \quad . \tag{1.3}$$

where

V_o = initial velocity transient (in./s),
ℓ = pipe length (in),
p = internal pressure (lb/in.^{2}),
c_1 = influence coefficient which transforms the initial velocity into a bending stress [(lb/in.^{2})/(in./s)],
c_2 = influence coefficient which transforms the pipe length into a bending stress [(lb/in.^{2})/in.],
c_3 = influence coefficient which transforms the internal pressure into an axial membrane stress [(lb/in.^{2})/(lb/in.^{2})],
S_m = stress intensity limit,
ϕ = resistance factor which reflects the stress categories in the load combination,
$\gamma_1, \gamma_2, \gamma_3$ = the load factors which correspond to V_o, ℓ, and p respectively.

2. We now evaluate the influence coefficients, which take the form

$$c_1 = \frac{4}{\pi}\left[\frac{E}{g}\left(4\Delta_s + \frac{D}{t}\Delta_w\right)\right]^{1/2} \cdot f(\ell)_{max} \quad . \tag{1.4a}$$

$$c_2 = \ell\left[\frac{\Delta_s}{D} + \frac{\Delta_w}{4t}\right] \quad . \tag{1.4b}$$

$$c_3 = \frac{D}{4t} \quad . \tag{1.4c}$$

where

E = modulus of elasticity of pipe material,
g = acceleration due to gravity,
Δ_s = density of pipe material,
Δ_w = density of fluid,
t = pipe thickness,

D = pipe diameter,
$f(\ell)$ = a factor which reflects the influence of all harmonics on the magnitude of the moment.

3. For an assumed set of load factors, we now solve the design format equation for the pipe thickness t that results in a stress equal to the factored code allowable limit stress. For this calculation the following values were used:

E = 30,000,000 lb/in.^{2},
g = 386 in./s^2,
Δ_s = 0.3 lb/in.^{3},
Δ_w = 0.026 lb/in.^{3},
S_m = 15,600 lb/in.^{2},
ϕ = 1.5 for combined primary membrane and bending stress,
p = 2200 lb/in.^{2},
V_o = 12 in./s.

4. Assume initial values for load factors, say, $\gamma_1 = \gamma_2 = \gamma_3 = 1$, and determine the thickness of pipe such that Eq. 1.3 is satisfied.

5. Once the thickness of the pipe has been determined, we have a design which can be analyzed probabilistically. The limit state probability is expressed by

$$P_F = 1 - \Phi(\beta) \quad , \tag{1.5}$$

where Φ is the Gaussian or normal distribution function for argument β, which can be expressed as

$$\beta = \frac{\bar{R} - \bar{Y}}{\left(\sigma_R^2 + \sigma_Y^2\right)^{1/2}} \tag{1.6}$$

where

$\overline{R}$ = the mean resistance,
σ_R^2 = the variance of the resistance,
$\overline{Y}$ = the mean response,
σ_Y^2 = the variance of the response.

The expressions for $\overline{Y}$ and σ_Y^2 are as follows:

$$\overline{Y} = c_1\overline{V}_o + c_2\overline{\ell} + c_3\overline{p} \quad . \tag{1.7}$$

and

$$\sigma_Y^2 = \left(\frac{\partial Y}{\partial V_o}\right)^2\sigma^2(V_o) + \left(\frac{\partial Y}{\partial \ell}\right)^2\sigma^2(\ell) + \left(\frac{\partial Y}{\partial p}\right)^2\sigma^2(p) \quad . \tag{1.8}$$

The values assumed for determining component limit state probability are:

$\overline{R}$ = 30,000 lb/in.^{2},
$\sigma_R = 0.05\overline{S}$ = 1500 lb/in.^{2},
$\sigma(V_o) = 0.10V_o$ = 1.2 in./s,
$\sigma(\ell)$ = 0.01,
$\sigma(p)$ = 285 lb/in.^{2}

6. Now specify a target limit state probability P_T for the pipe component, and note the difference between this probability and the computed limit state probability P_F. Since we want the load factors to represent as broad a class of pipe configurations as possible, we perform the calculation for the combination of eight pipe diameters and five lengths displayed in Table 1.1. Associated with each unique combination of length and diameter is a pipe thickness that limits the stress to the code allowable for the applied load. A limit state probability is also associated with each combination of length and diameter. For this array we set up an objective function,

$$f(\gamma) = \sum_{i=1}^{8}\sum_{j=1}^{5}\left[\frac{\log P_{F_{ij}} - \log P_T}{\log P_T}\right]^2 \quad . \tag{1.9}$$

APPLICATIONS OF LOAD COMBINATION METHODOLOGY TO NUCLEAR COMPONENTS

M. K. Ravindra
Sargent & Lundy, Engineers

The objective of the probabilistic approach to load combination development is to consistently account for the stochastic behavior of loads and the uncertainties in component strengths. Consistency is attained if the components are designed to achieve the target limit state probabilities, which in turn would keep the plant risk at acceptably low levels. For practical reasons, this goal of probabilistic design has to be achieved within the deterministic framework of current design criteria. The design format chosen as the most appropriate for routine design of components is the familiar "load and resistance factor design," (LRFD) formula used in the NRC Standard Review Plan (Sec. 3.8.4) and in the industry codes, such as ACI 349 and the proposed AISC Nuclear Specifications. The format is deterministic in that all the factors are specified. The general formula in the LRFD format is

$$\phi R_n \geq \gamma_1 c_1 X_1 + \gamma_2 c_2 X_2 + \ldots \qquad (1)$$

where

R_n = nominal resistance of the component

ϕ = resistance factor

γ_i = load factor on the ith load

c_i = influence coefficient for the ith load

X_i = design intensity of the ith load

This formula states that the factored nominal resistance of the component should exceed the sum of the factored nominal load effect acting in a combination. Herein, the term "load effect" means imposed stress, response, moment, etc; the resistance is correspondingly strength, moment carrying capacity, etc. The factors ϕ and γ_i reflect the uncertainties in predicting the resistance, loads, and target limit state probability of the component. The nominal values of loads and resistances are selected based on specified probabilities of nonexceedance or on the current industry practice. In these load combinations, the responses may be combined using the SRSS or absolute sum procedures.

For each limit state, there will be a set of load combinations to be considered. Current design practice is to find the maximum response from all the relevant load combinations and to design the component's resistance to exceed this maximum response. Therefore, such sets of load combination formulas are to be considered for limit states of functional failure, minor damage, and catastrophic failure. Similar sets of design equations are provided for different components. This is the complete process of designing components. It closely follows current practice. The ASME code is similar in philosophy though different in appearance. An advantage of having factors on the resistance and loads instead of on the resistance only as in the ASME code is the greater flexibility in the format. Because of this flexibility, consistent component

reliabilities for different limit states and design situations can be achieved. However, in the initial stages, the option of retaining the ASME code format is not ruled out. The impact of the load combination methodology would then be seen only in the revised allowable stresses for different service levels corresponding to the specified target limit state probabilities.

The derivation of load and resistance factor values is described briefly below.

First, a design format (i.e., nominal loads, nominal resistances, load combinations, methods of analysis, and response combination procedures) is selected. For an assumed set of load and resistance factors, a representative component is designed. This component is acted upon by random loads singly or in combination; there is also some uncertainty in the component's resistance. We need to evaluate the component limit state probability, P_F. If this probability is equal to the target value P_T, then our choice of load and resistance factors is very fortunate. It may not be so for other components in the particular class of components. Hence, a measure of closeness is needed to judge if the component target limit state probability is acheived using these load combinations. This measure, called the objective function Ω, is an expectation over the set of all design situations for the particular class of components. By minimizing the objective function, the optimal values of ϕ and γ are obtained.

As an illustration of this procedure, the values of ϕ to be used in an ASME-type design format for ESW piping are derived for target probabilities of reaching limit states of pipe rupture, restraint buckling, and unacceptable acceleration.

LOAD AND RESISTANCE FACTOR DESIGN FORMAT

$$\phi R_n \geq \gamma_1 c_1 X_1 + \gamma_2 c_2 X_2 + \gamma_3 c_3 X_3 + \cdots$$

R_n = nominal resistance of the component

ϕ = resistance factor

γ_i = load factor

c_i = influence coefficient

X_i = design load intensity

DEVELOPMENT OF LOAD COMBINATIONS

Design Format

A. Enumeration of Loads; Nominal Loads

B. Design Load Combinations

C. Method of Analysis

D. Dynamic Response Combination Procedure

E. Procedures for Safety Checking/ Proportioning Component

F. Load and Resistance Factors

A. Loads

Loads	Nominal Loads
Weight, W	Weight, W
Seismic, E	Seismic, OBE, SSE
Hydraulic Transient, HTR	Hydraulic Transient $(HTR)_N$
Thermal, TH	Design Pressure, P_d
Pressure, P	Thermal Load, TRNG

B. Load Combinations

$$\phi_{11}R_1 \geq \gamma_P c_P P_d + \gamma_W c_W W$$

$$\phi_{12}R_1 \geq \gamma_P c_P P_d + \gamma_W c_W W + \gamma_{E_{10}} c_{E_o} (OBE)$$

$$\phi_{12}R_1 \geq \gamma_P c_P P_d + \gamma_W c_W W + \gamma_{H_0} c_H (HTR)_N$$

$$\phi_{13}R_1 \geq \gamma_P c_P P_d + \gamma_W c_W W + \gamma_{TH} c_{TH} (TRNG)$$

$$\phi_{14}R_1 \geq \gamma_P c_P P_d + \gamma_W c_W W + \gamma_{E_{11}} c_{E_o} (OBE) + \gamma_{H_1} c_H (HTR)_N$$

$$\phi_{14}R_1 \geq \gamma_P c_P P_d + \gamma_W c_W W + \gamma_{E_{12}} c_{E_S} (SSE)$$

$$\phi_{21}R_{21} \geq \gamma_{E_{21}} c_{E_{oA}} (OBE)$$

$$\phi_{21}R_{21} \geq \gamma_{H_A} c_{H_A} (HTR)_N$$

$$\phi_{22}R_{22} \geq \gamma_{E_{22}} c_{E_{SA}} (SSE)$$

$$\phi_{22}R_{22} \geq \gamma_{H_{A1}} c_{H_A} (HTR)_N + \gamma_{E_{23}} c_{E_{oA}} (OBE)$$

C. Analysis

Linear elastic; earthquake response analysis using response spectrum method; hydraulic transients analyzed using time history integration.

D. Response Combination

SRSS

E. Method of Proportioning

Satisfaction of stress equations, capacity tables, limiting acceleration values.

LOAD AND RESISTANCE FACTORS

Find $(\underset{\sim}{\phi}, \underset{\sim}{\gamma})$ such that

$$\Omega(\underset{\sim}{\phi}, \underset{\sim}{\gamma}) = \sum_{\underset{\sim}{\omega} \in \mathcal{D}} f(\underset{\sim}{\omega}) \left[\frac{\log P_F(\underset{\sim}{\omega}) - \log P_T}{\log P_T} \right]^2$$

and

$$P[P_F > P_{f_n}] < 0.05$$

ALTERNATE FORMATS

1. ASME Design Format
2. Different Response Combination Procedures
3. Event Decoupling
4. Use of Multiple Load Levels

ASME DESIGN FORMAT

$$\phi_{11} S_h \geq c_P P_d + c_W W$$

$$\phi_{12} S_h \geq c_P P_d + c_W W + C_{E_o}(OBE)$$

$$\phi_{12} S_h \geq c_P P_d + c_W W + C_H (HTR)_N$$

$$\phi_{13} S_h \geq c_P P_d + c_W W + C_{TH}(TRNG)$$

$$\phi_{14} S_h \geq c_P P_d + c_W W + C_{E_o}(OBE) + C_H (HTR)_N$$

$$\phi_{14} S_h \geq c_P P_d + c_P W + C_{E_S}(SSE)$$

$$\phi_{21} R_{21} \geq C_{E_{oA}}(OBE)$$

$$\phi_{21} R_{21} \geq C_{H_A} (HTR)_N$$

$$\phi_{22} R_{22} \geq C_{H_A} (HTR)_N + C_{E_{oA}}(OBE)$$

$$\phi_{22} R_{22} \geq C_{E_{SA}}(SSE)$$

OPTIMAL VALUES OF RESISTANCE FACTORS
FOR P_T (10^{-12}, 10^{-5}, 10^{-5}/yr)

$\phi_{11} = 0.72$

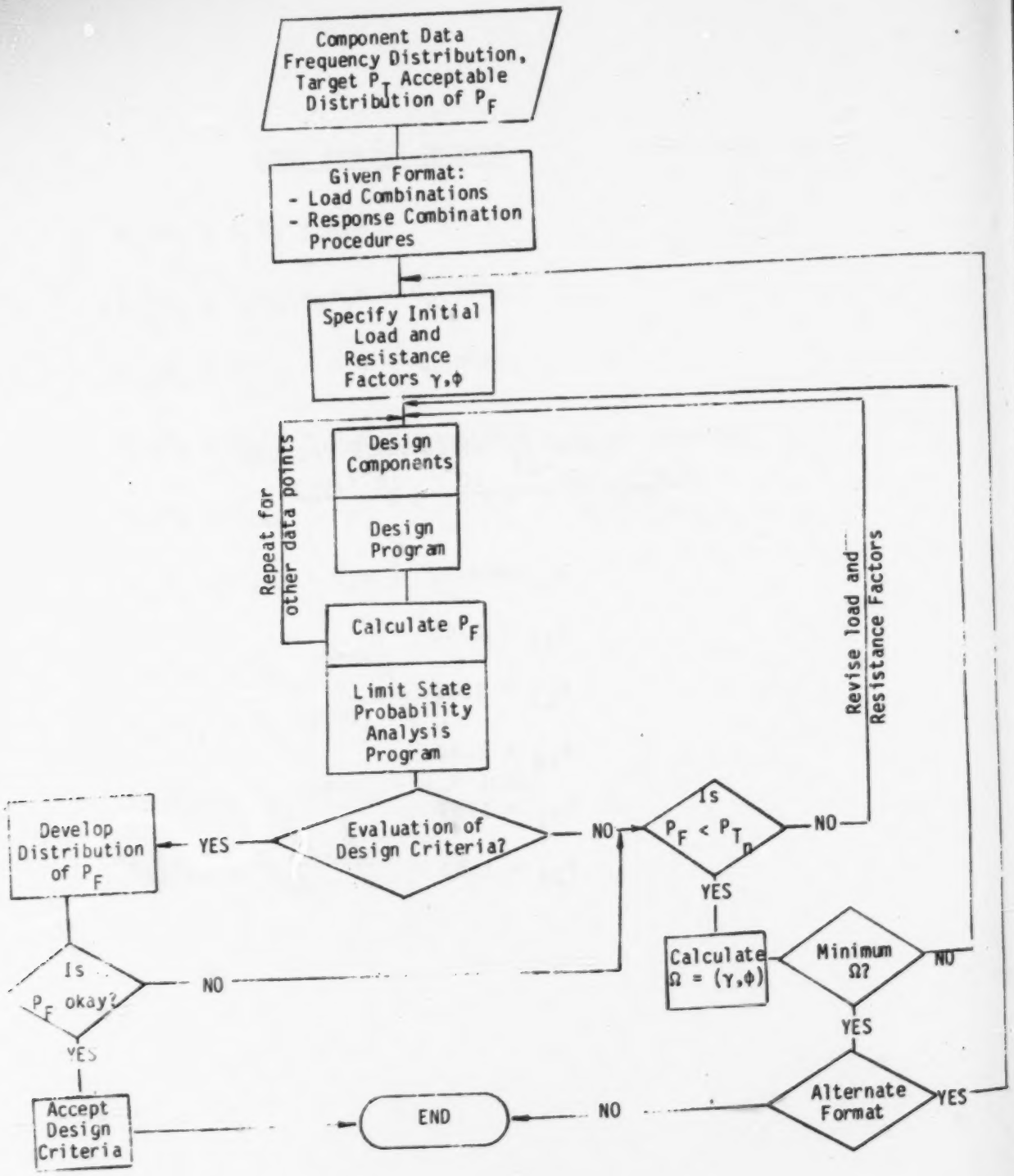
$\phi_{12} = 2.00$

$\phi_{13} = 2.00$

$\phi_{14} = 1.60$

$\phi_{21} = 1.89$

$\phi_{22} = 1.52$



LOAD COMBINATION METHODOLOGY

FUTURE WORK

1. Data Collection:
   a. Develop a format
   b. Loads
   c. Component Fragilities
   d. Influence Coefficient Ratio Distribution
2. Response Analysis
3. Parametric Studies
4. Efficient Algorithms for Optimization
5. Evaluate Current Design Criteria

USC/NRC Contract
NRC-04-76-262
NONLINEAR SYSTEM MODELING

IDENTIFICATION AND MODELING OF NONLINEAR SYSTEMS

by

S.F. Masri

School of Engineering
University of Southern California
Los Angeles, CA 90007
U.S.A.

SUMMARY

The identification and modeling of multidegree-of-freedom dynamic systems through the use of experimental data is a problem of considerable importance in the area of structural dynamics. Indicative of the wide range of applicability of this subject is a recent survey article which contains over 130 references related to system and parameter identification.

System identification techniques can be classified on the basis of their search space: (a) parametric methods that search in parameter space, and (b) nonparametric methods that search in function space. Parametric methods are effective when adequate information about the structure of the model is available and only the parameter magnitudes are to be determined. Nonparametric methods, on the other hand, produce the "best" functional representation of the system without à priori assumptions about the system model.

In many practical dynamic problems, such as mechanical equipment in nuclear power plants, the structure of the model is by no means clear.

As a result, an increasing amount of attention has recently been devoted to nonparametric identification methods such as the ones that use the Volterra-series or Wiener-kernel approach.

However, the traditional nonparametric identification techniques have their own problems. Some of these include restrictions on the nature of the dynamic systems to be identified and on the input signal that can be used. Furthermore, when dealing with systems that incorporate commonly encountered nonlinearities, such as polynomial nonlinearities, the evalaution of higher-order terms requires a prohibitive amount of computational effort, coupled with very demanding storage requirements.

This paper presents a relatively simple and straightforward approach to the identification and modeling of a broad class of nonlinear multi-degree-of-freedom dynamic systems. Such a class of models encompasses many practical engineering problems including certain equipment systems, tall buildings, and other similar structures, all of which are governed by equations of motion that are characteristic of chain-like systems.

The method requires information regarding the system displacement, velocity, and acceleration to determine, by means of regression techniques involving the use of two-dimensional orthogonal functions, an approximate expression for the surface of each of the system restoring forces in terms of the corresponding system state variables.

The main features of this method are:

(1) Practically any type of nonlinear system characteristics can be identified. Virtually exact answers can be obtained for nonhysteretic

systems even if the nonlinearity involves cross-product terms that depend on displacement and velocity. Hysteretic systems, which pose problems for conventional nonparametric identification techniques, can be easily handled and reasonably estimated by the present method since they are not treated any differently from other arbitrary nonlinearities.

(2) Virtually any type of probing signal can be used for identification; i.e., random signals (stationary and nonstationary) and swept-sine signals are equally suitable.

(3) Very modest amounts of computer time are needed to implement the method.

(4) Computer storage requirements are extremely compact for the characterization of arbitrarily nonlinear systems.

(5) Fast convergence can be achieved with very few terms in the series expansion even in the case of hysteretic systems.

(6) Noise pollution of the data has a minimal effect on the identification results obtained by this technique.

The method is applied to a three-degree-of-freedom model of a nuclear power plant valve, and is shown by means of numerical "experiments" that it is very efficient and can be utilized under realistic conditions to pinpoint localized nonlinearities of arbitrary types wherever they occur in the system.

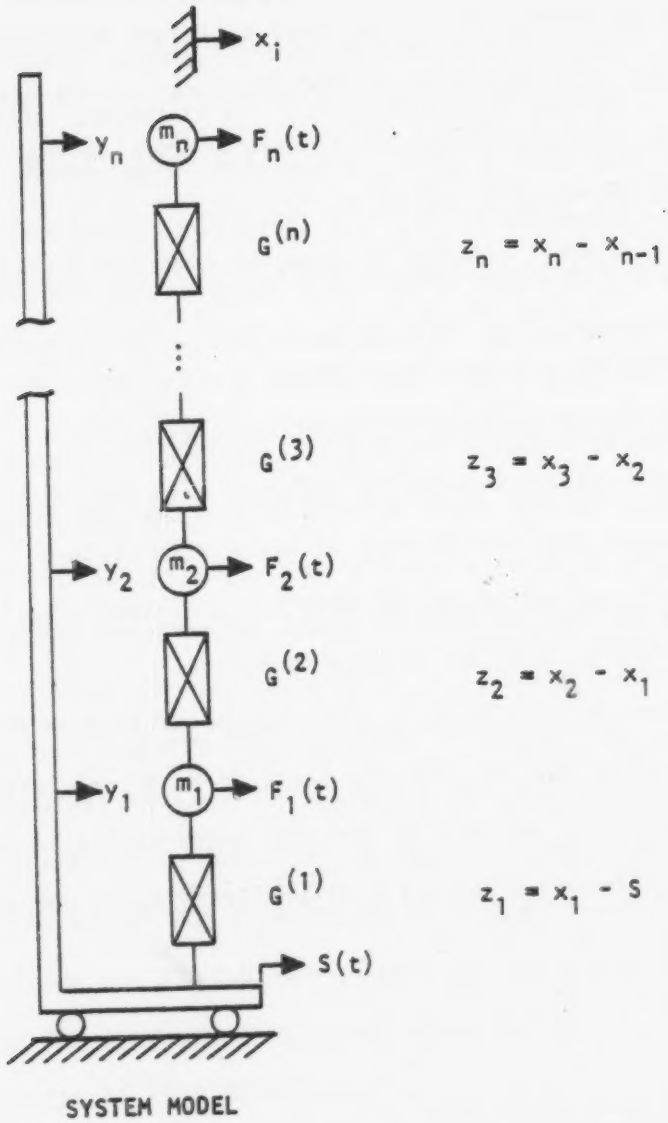
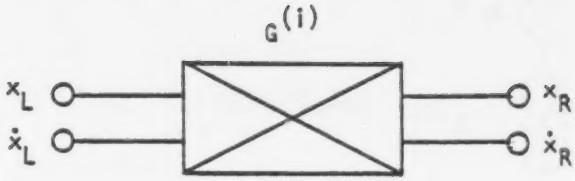


FIGURE 1. MODEL OF NONLINEAR MULTIDEGREE-OF-FREEDOM CHAIN-LIKE SYSTEM



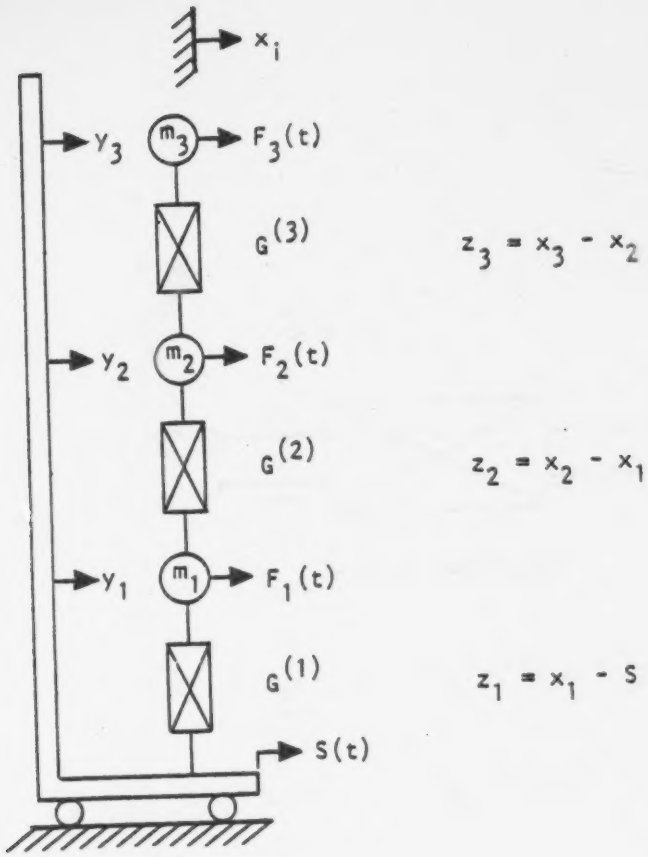
NONLINEAR ELEMENT

$$G^{(i)} = G^{(i)}\,(\underset{\sim}{p}^{(i)},\ x_L,\ x_R,\ \dot{x}_L,\ \dot{x}_R)$$

$$= G^{(i)}\,(\underset{\sim}{p}^{(i)},\quad z\ ,\quad \dot{z}\ )$$

where $z = x_R - x_L$

FIGURE 2. REPRESENTATION OF ARBITRARY NONLINEAR ELEMENTS



(a) System model

$$G^{(i)}(z, \dot{z}) = p_1^{(i)} z + p_2^{(i)} \dot{z} + p_3^{(i)} z^3$$

ELEMENT CHARACTERISTICS

$G^{(i)}$

i	1	2	3
m_i	12.0	12.0	12.0
p(1, i)	7214.0	14900.0	14900.0
p(2, i)	21.0	60.0	60.0
p(3, i)	72.14	0.0	-223.0

(b) System characteristics

FIGURE 3. EXAMPLE OF NONLINEAR 3 DOF SYSTEM

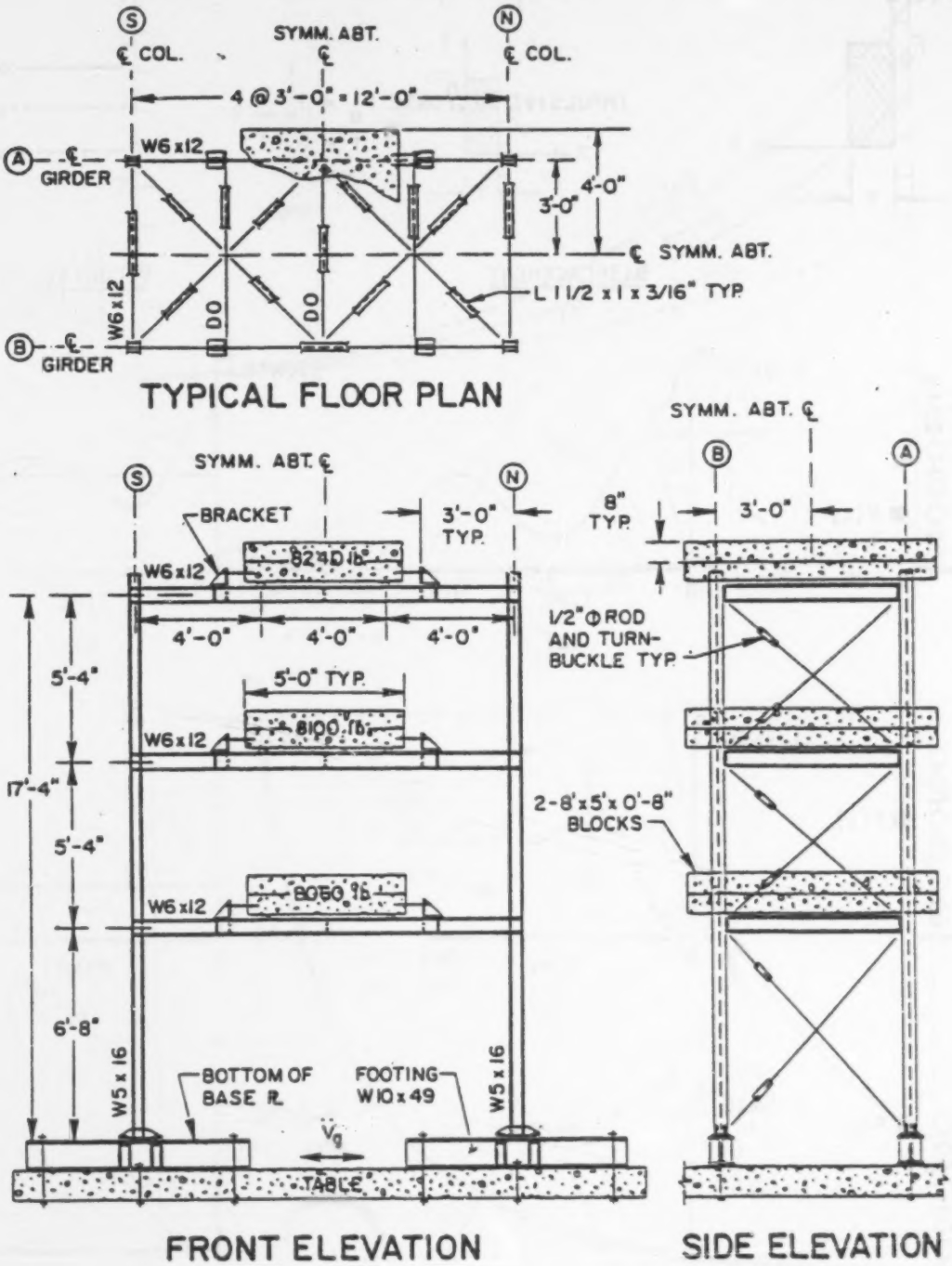


FIGURE 4. PLAN AND ELEVATIONS OF THE TEST STRUCTURE [34]

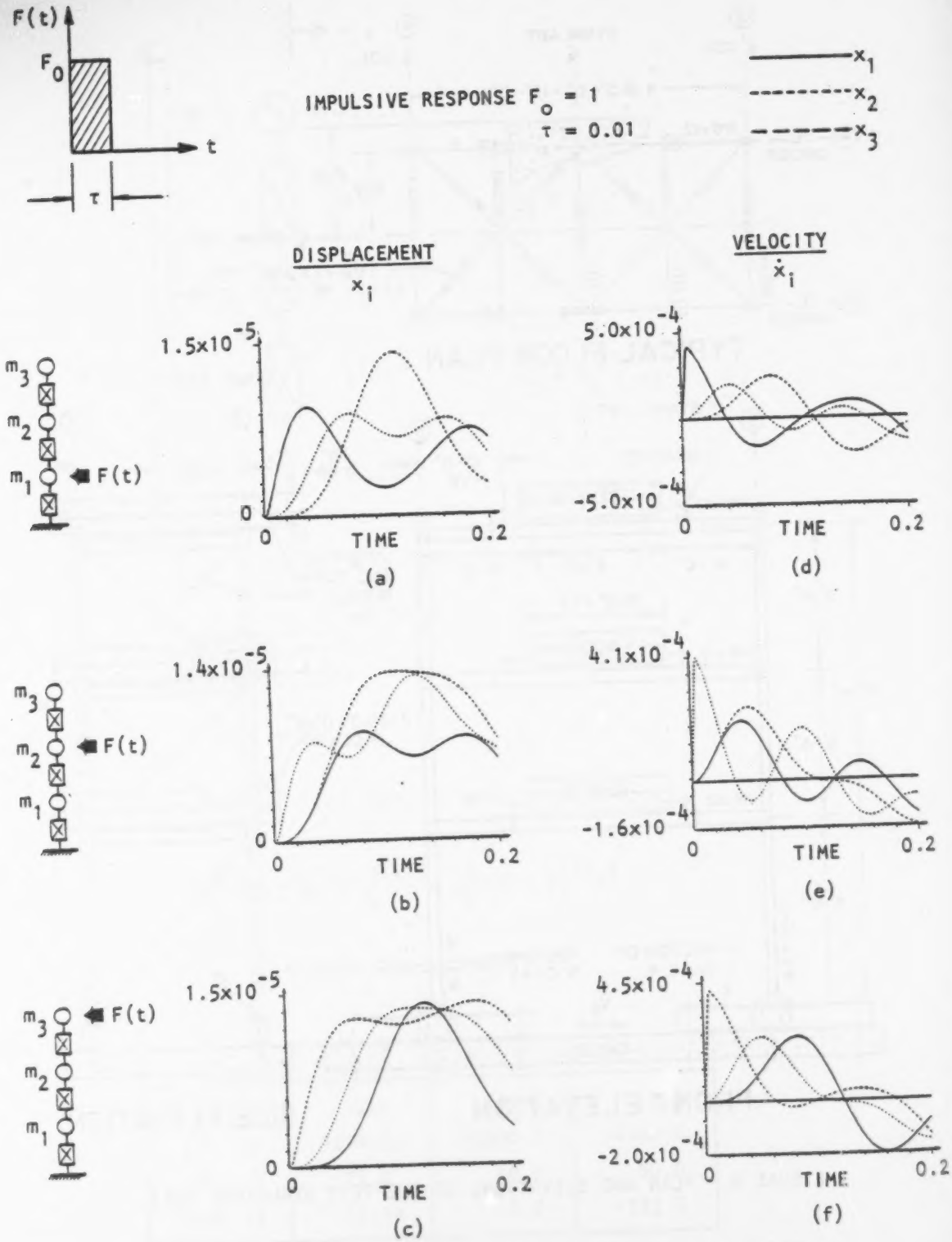


FIGURE 5. IMPULSIVE RESPONSE OF A NONLINEAR 3 DOF SYSTEM

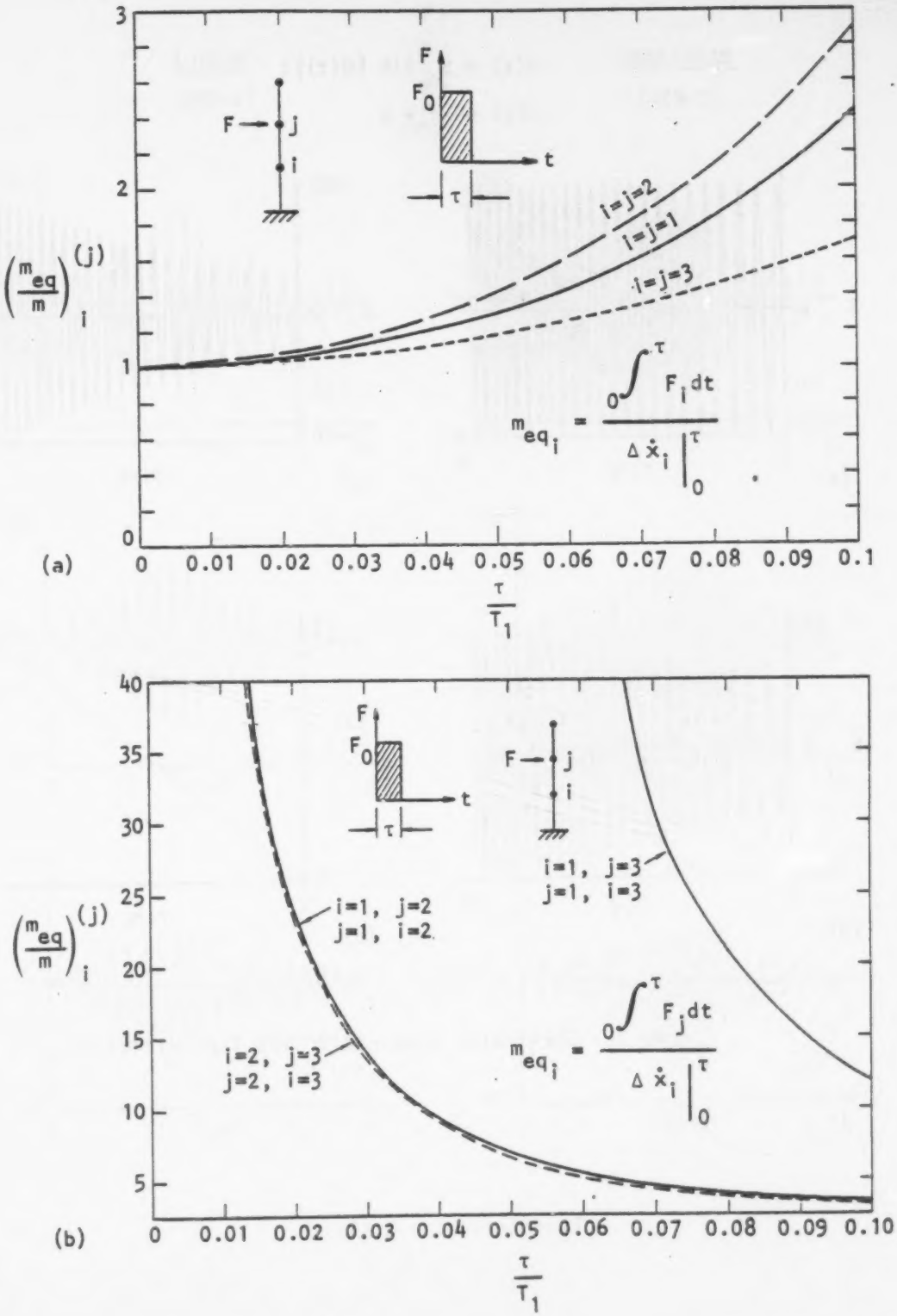


FIGURE 6. IMPULSIVE RESPONSE OF A NONLINEAR 3 DOF SYSTEM

$$S(t) = S_o \text{ SIN } [\Omega(t)]\, t$$

$$\Omega(t) = at + b$$

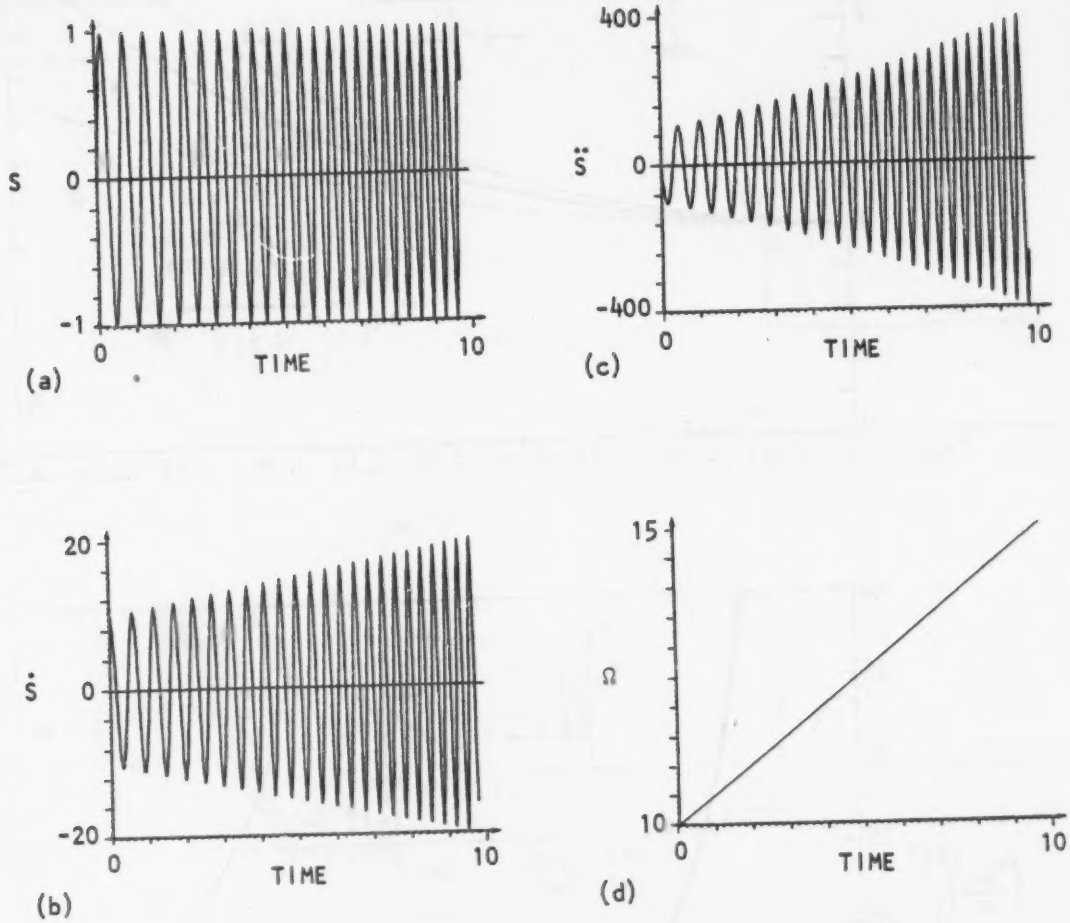


FIGURE 7. SWEPT-SINE BASE EXCITATION TIME HISTORIES

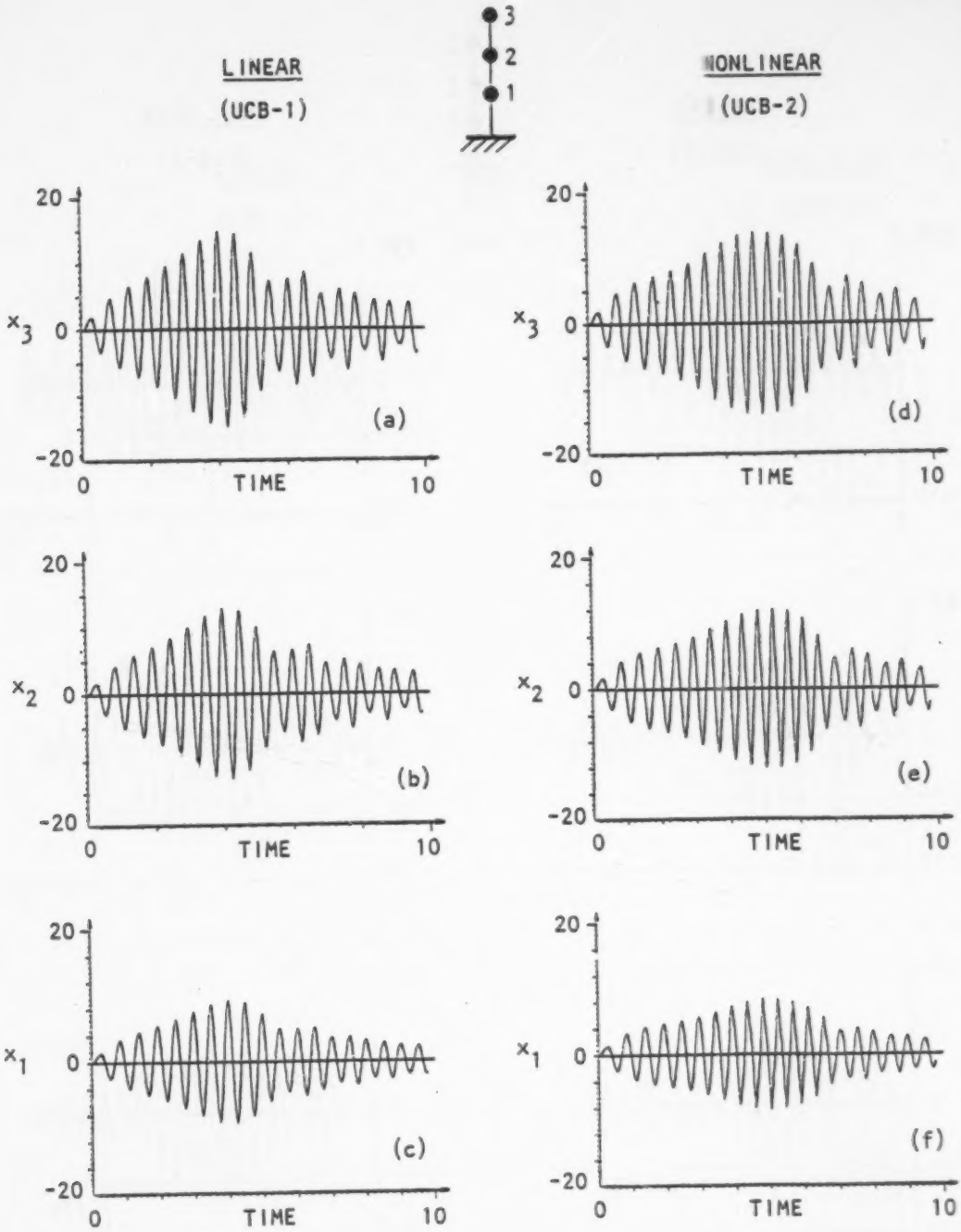


FIGURE 8. DISPLACEMENT

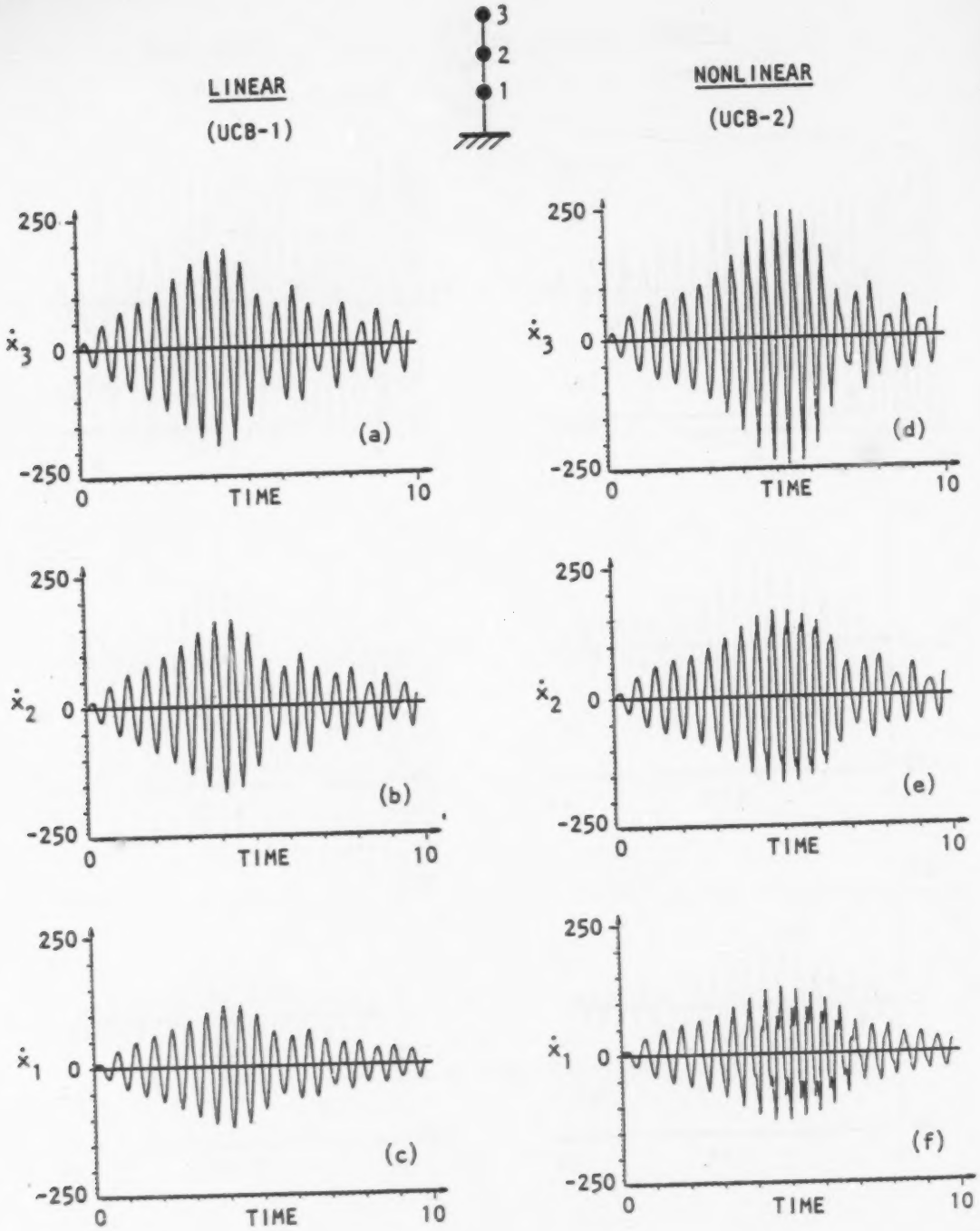


FIGURE 9. VELOCITY

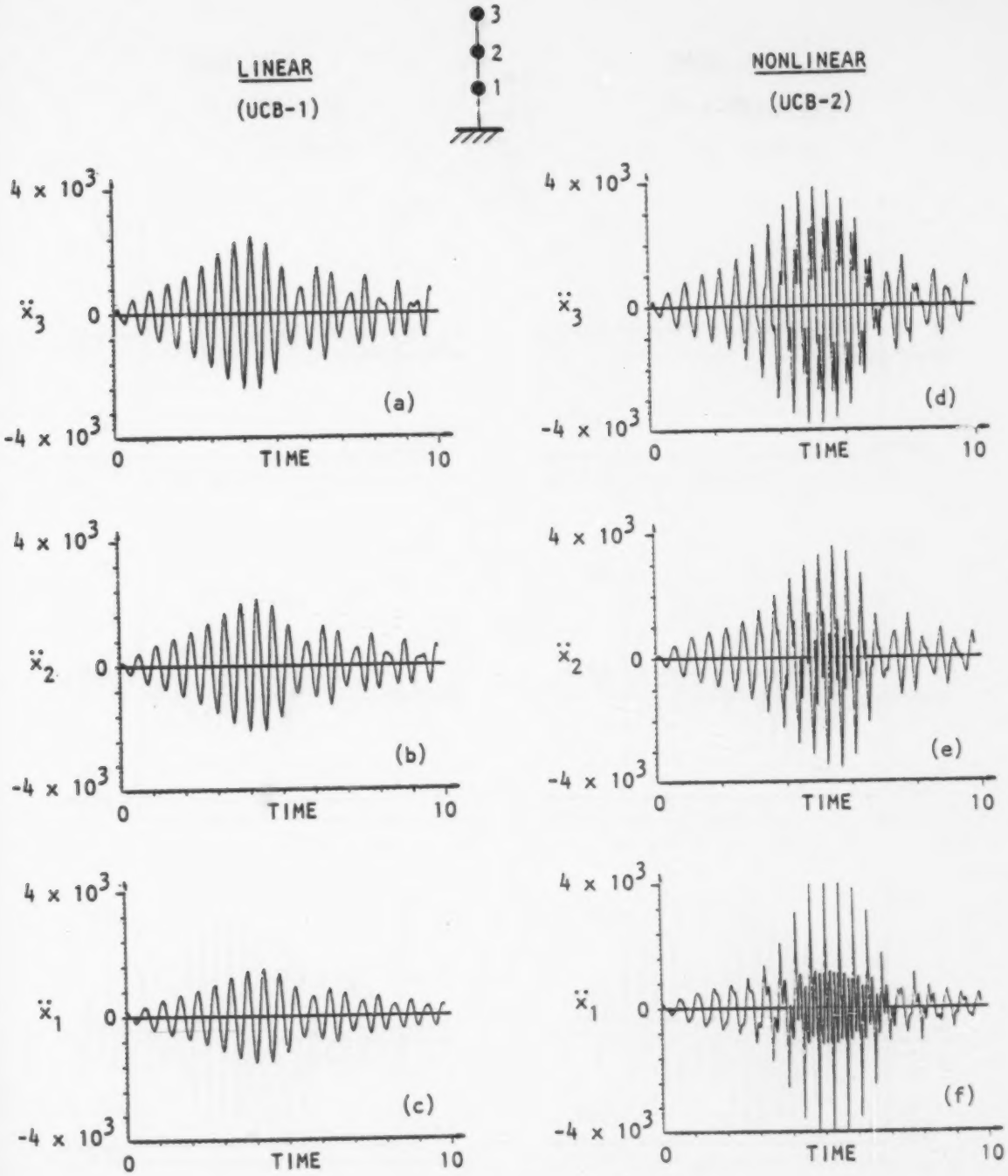


FIGURE 10. ACCELERATION

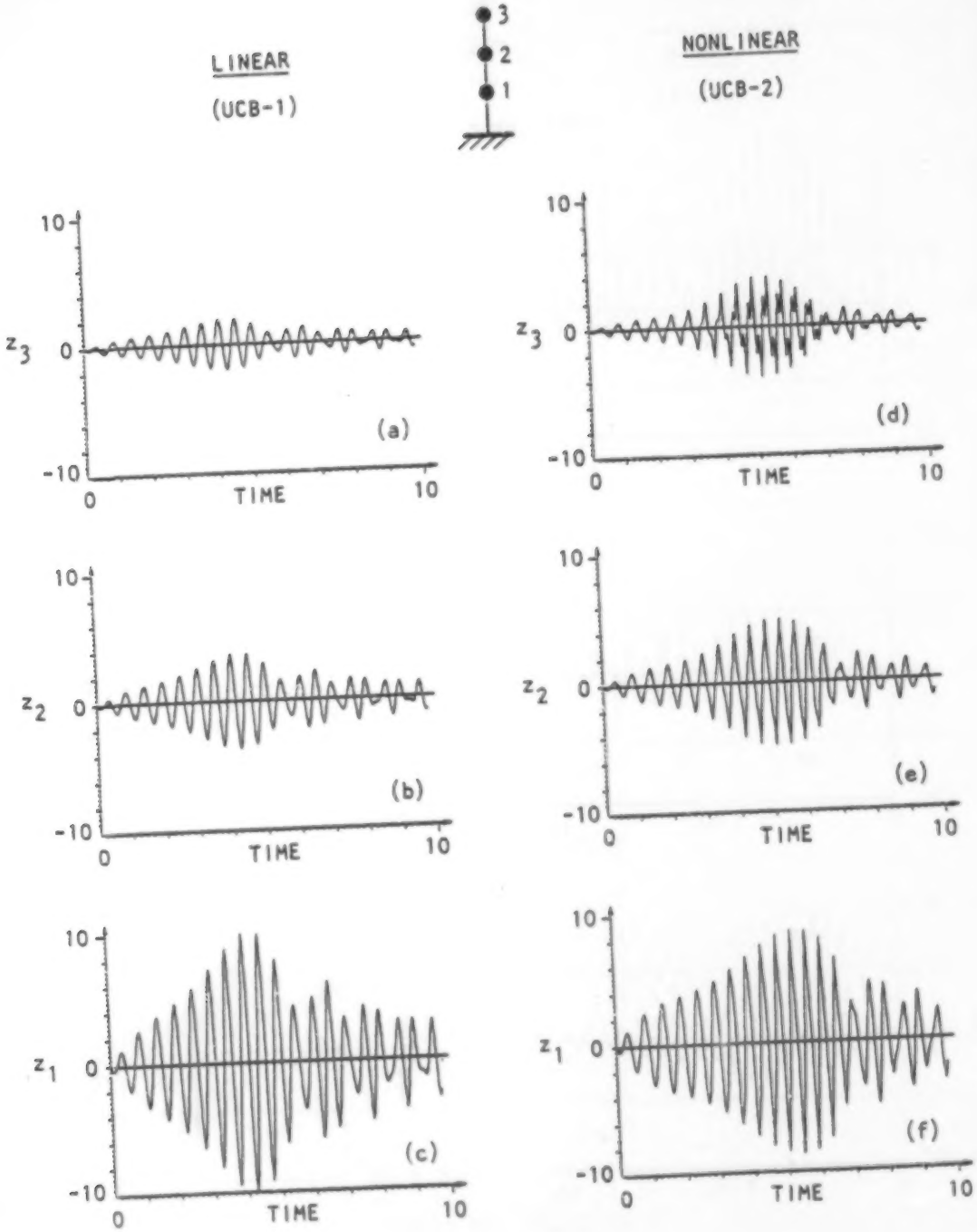


FIGURE 11. RELATIVE DISPLACEMENT

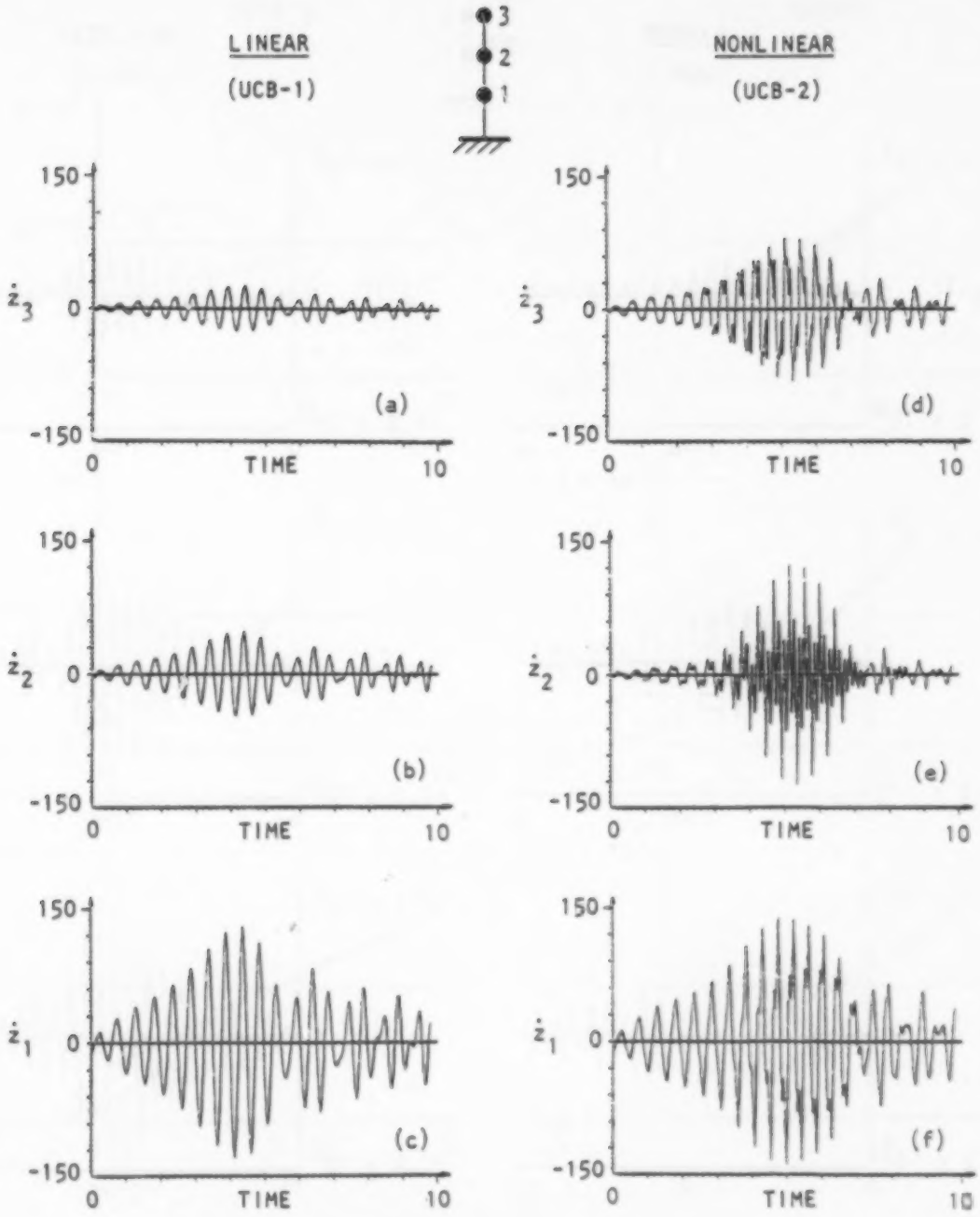


FIGURE 12. RELATIVE VELOCITY

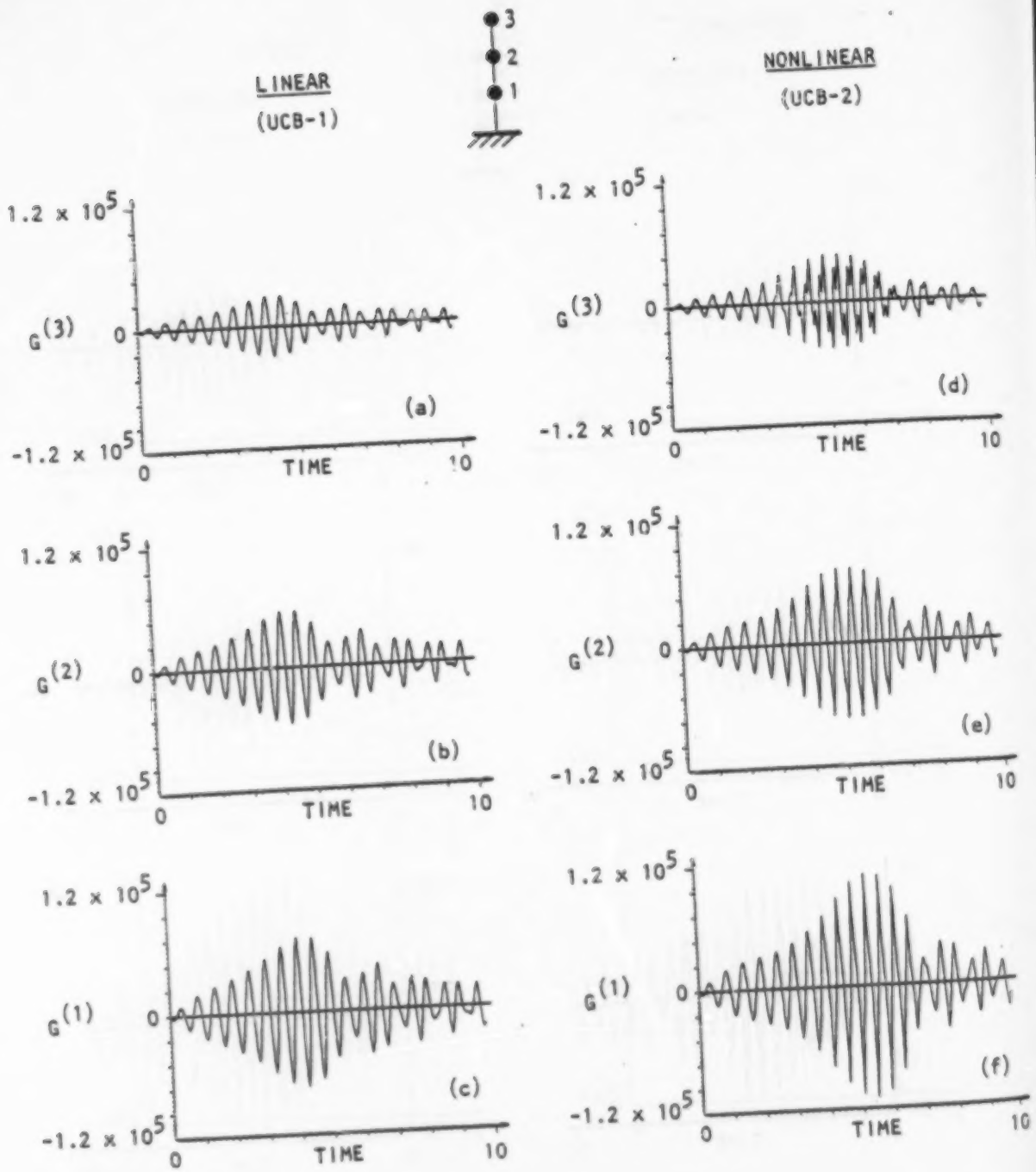


FIGURE 13. RESTORING FORCE

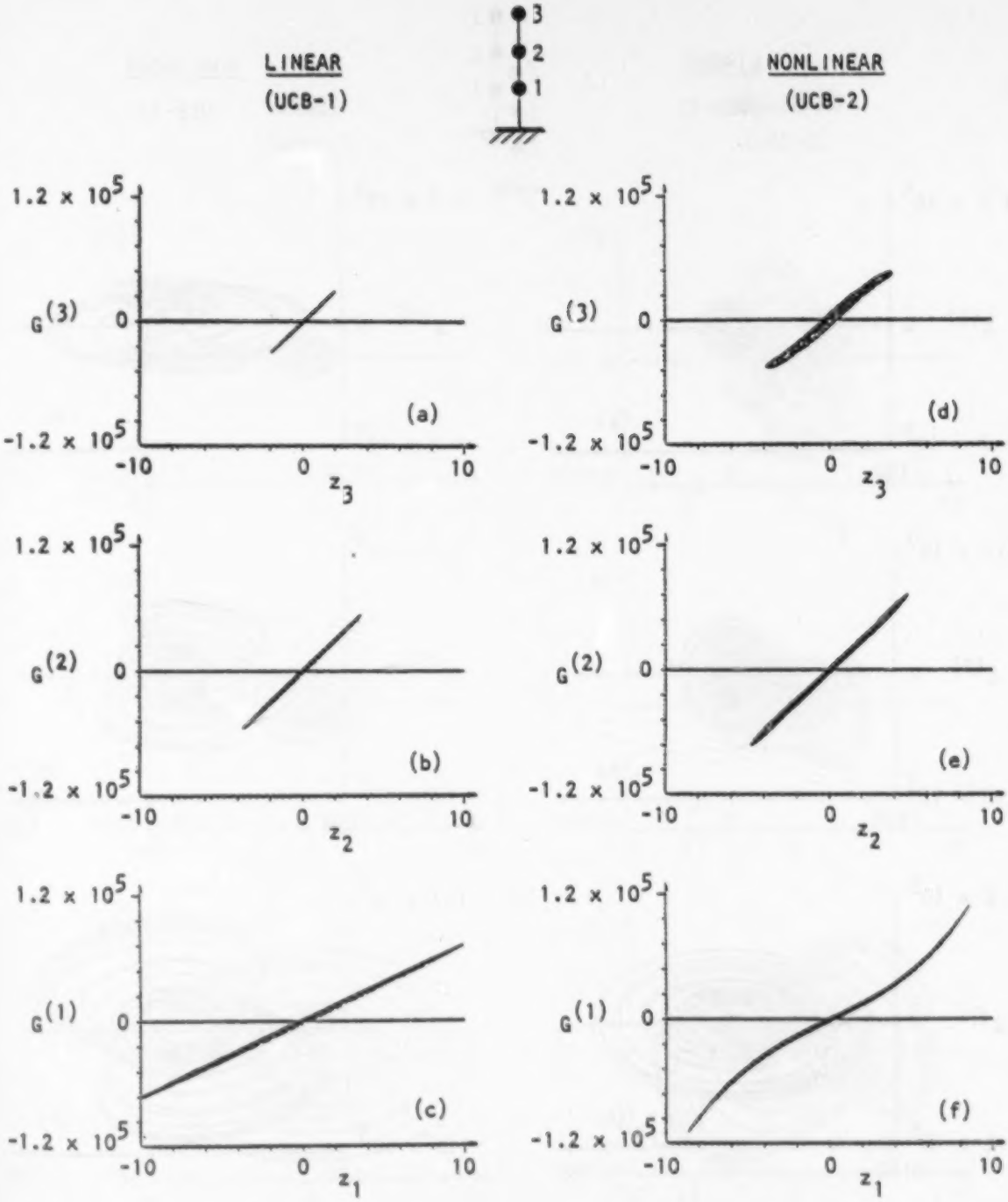


FIGURE 14. STATE VARIABLES: Z VS. FORCE

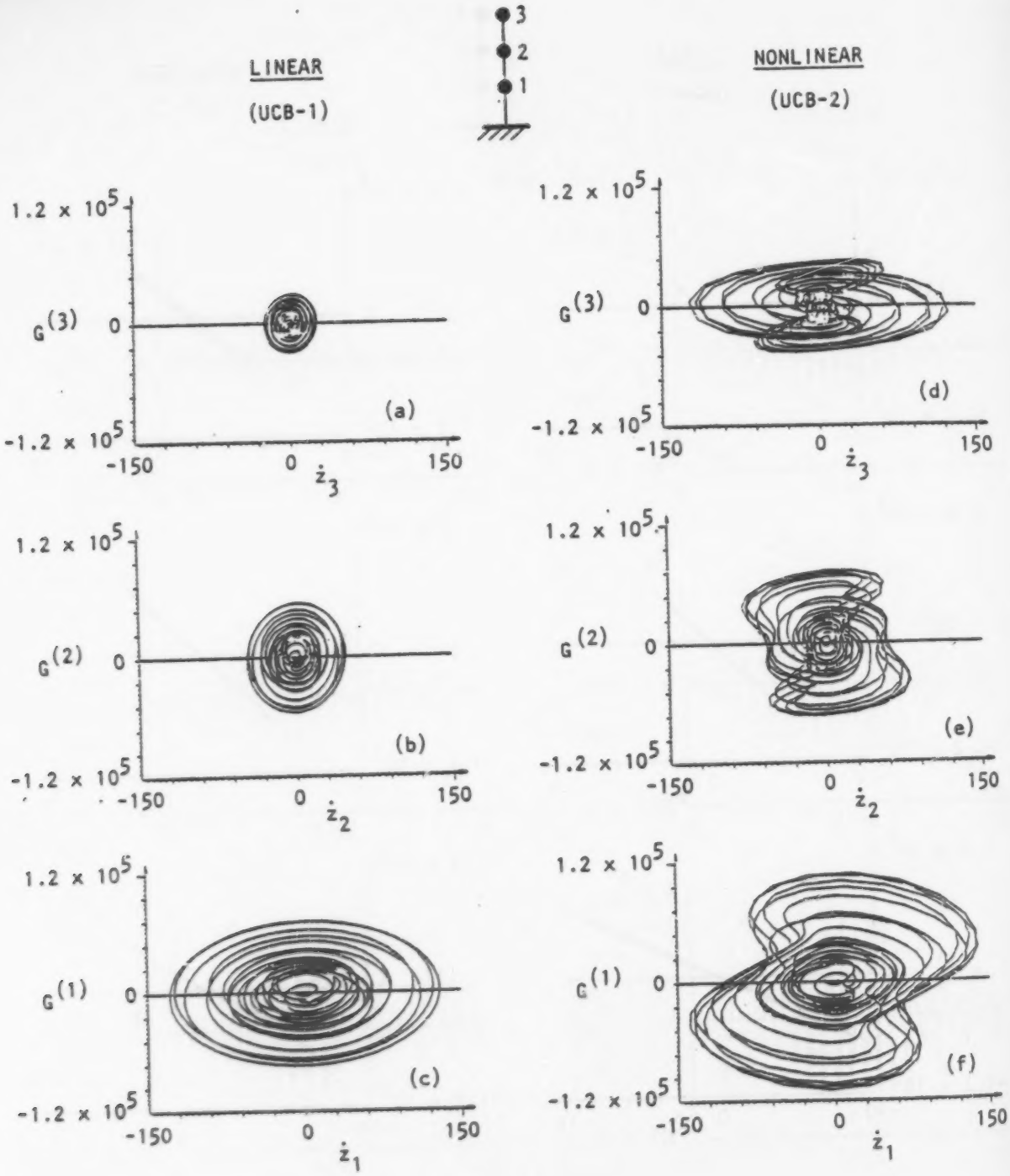


FIGURE 15. STATE VARIABLES: $\dot{Z}$ VS. FORCE

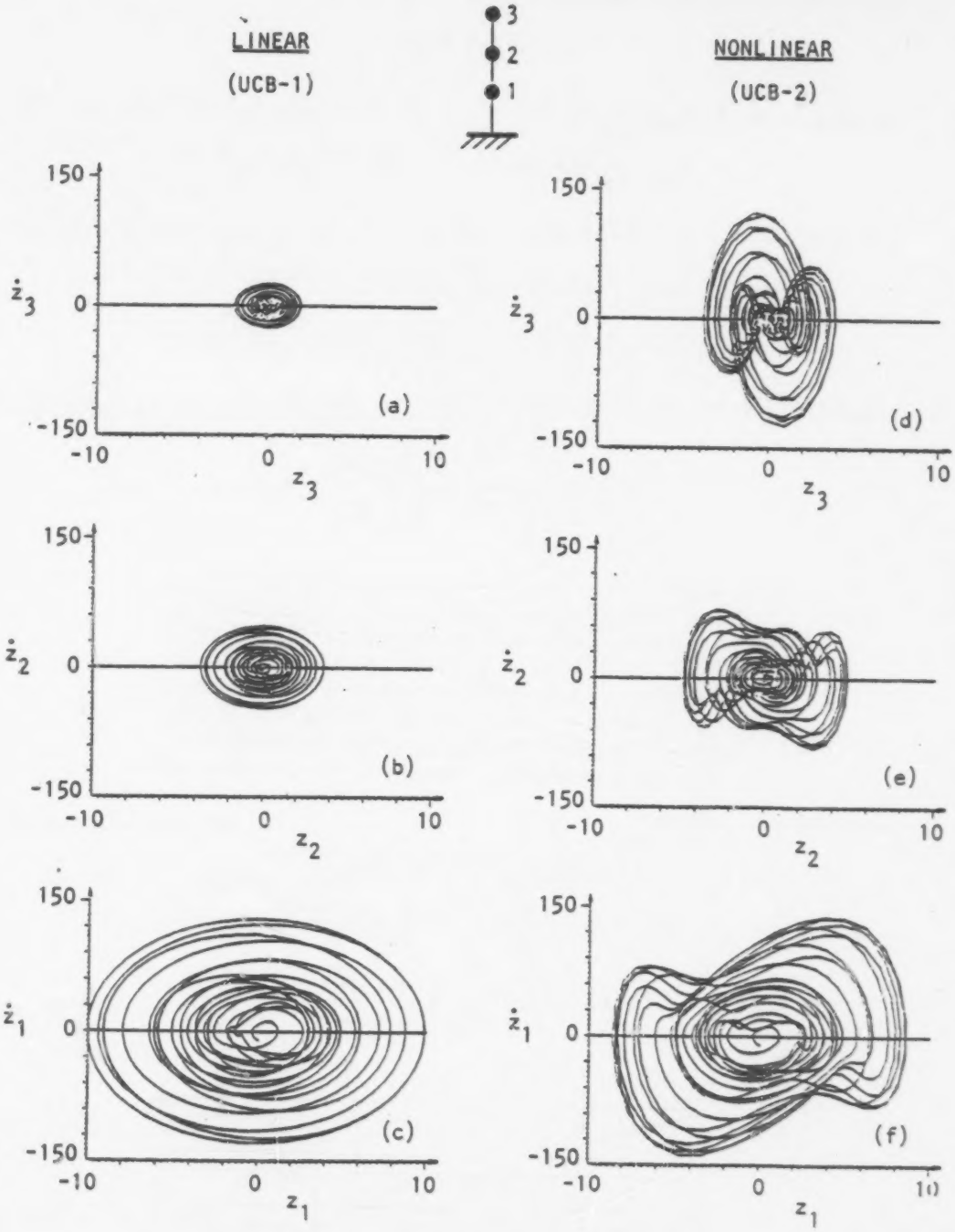


FIGURE 16. STATE VARIABLES: Z VS. $\dot{Z}$

Actual System: LINEAR (Element 3)

EXACT $G(z,\dot{z})$: $1.49 \times 10^4 z + 60.0\dot{z}$

Identification Excitation: $S(t) = S_o \sin(at^2 + bt)$

$S_o = 1.0$

Motion Range: $z_{min} = -1.97$ $\quad z_{max} = 1.93$

$\dot{z}_{min} = -25.20$ $\quad \dot{z}_{max} = 25.45$

1D Fit:

$$\hat{G}(z,\dot{z}) = g(z) + h(\dot{z})$$

$$\hat{g}(z) = -8.18 \times 10^2 T_o(z') + 5.68 \times 10^4 T_1(z')$$

$$\hat{h}(\dot{z}) = 4.13 \times 10^2 T_o(\dot{z}') + 2.96 \times 10^4 T_1(\dot{z}')$$

Coefficient of 2D Chebyshev series:

$$G(z,\dot{z}) = \sum_{i=0}^{m2} \sum_{j=0}^{n2} C_{ij} T_i(z') T_j(\dot{z}')$$

i \ m	$T_o(\dot{z}')$	$T_o(\dot{z}')$
$T_o(z')$	4.50	1.17×10^3
$T_1(z')$	2.91×10^4	1.69×10^{-1}

Polynomial series:

1D Fit:

$$g(z) = -2.5 \times 10^2 + 2.91 \times 10^4 z'$$

$$h(\dot{z}) = 2.67 \times 10^2 + 1.17 \times 10^3 \dot{z}'$$

2D Fit:

$$G(z,\dot{z}) = \sum_{\ell=0}^{m2} \sum_{m=0}^{n2} A_{\ell m} z'^{\ell} \dot{z}'^{m}$$

ℓ \ m	1	$\dot{z}'$
1	2.89×10^2	46.36
z'	1.49×10^4	3.41×10^{-3}

FIGURE 17.

Actual System: LINEAR (Element 2)

EXACT $G(z,\dot{z})$: $1.49 \times 10^4 z + 60.0\dot{z}$

Identification Excitation: $S(t) = S_o \sin(at^2 + bt)$

$S_o = 1.0$

Motion Range: $z_{min} = -3.68$ $\quad z_{max} = 3.60$

$\dot{z}_{min} = -47.00$ $\quad \dot{z}_{max} = 47.41$

1D Fit:

$$\hat{G}(z,\dot{z}) = g(z) + h(\dot{z})$$

$$\hat{g}(z) = -2.61 \times 10^3 T_o(z') + 1.97 \times 10^5 T_1(z')$$

$$\hat{h}(\dot{z}) = 1.74 \times 10^2 T_o(\dot{z}') + 1.98 \times 10^5 T_1(\dot{z}')$$

Coefficient of 2D Chebyshev series:

$$\hat{G}(z,\dot{z}) = \sum_{i=0}^{m2} \sum_{j=0}^{n2} C_{ij} T_i(z') T_j(\dot{z}')$$

i \ j	$T_o(\dot{z}')$	$T_o(\dot{z}')$
$T_o(z')$	-1.21×10^3	4.17×10^3
$T_1(z')$	5.43×10^4	-2.51

Polynomial series:

1D Fit:

$$g(z) = -5.44 \times 10^2 + 5.43 \times 10^4 z'$$

$$h(\dot{z}) = -6.86 \times 10^2 + 4.19 \times 10^3 \dot{z}'$$

2D Fit:

$$G(z,\dot{z}) = \sum_{\ell=0}^{m2} \sum_{m=0}^{n2} A_{\ell m} z'^{\ell} \dot{z}'^{m}$$

ℓ \ m	1	$\dot{z}'$
1	-6.57×10^2	88.27
z'	1.49×10^4	-1.46×10^{-2}

FIGURE 18.

Actual System: LINEAR (Element 1)

EXACT $G(z,\dot{z})$: $7214.0z + 21.0\dot{z}$

Identification Excitation: $S(t) = S_o \sin(at^2 + bt)$

$S_o = 1.0$

Motion Range: $z_{min} = -10.09$ $\quad z_{max} = 9.93$

$\dot{z}_{min} = -1.29 \times 10^2$ $\quad \dot{z}_{max} = 1.30 \times 10^2$

1D Fit:

$$G(z,\dot{z}) = g(z) + h(\dot{z})$$

$$g(z) = -6.51 \times 10^3 T_o(z') + 7.23 \times 10^5 T_1(z')$$

$$h(\dot{z}) = 1.05 \times 10^3 T_o(\dot{z}') + 6.85 \times 10^5 T_1(\dot{z}')$$

Coefficient of 2D Chebyshev Series:

$$G(z,\dot{z}) = \sum_{i=0}^{m2} \sum_{j=0}^{n2} C_{ij} T_i(z') T_j(\dot{z}')$$

i \ j	$T_o(z')$	$T_1(\dot{z}')$
$T_o(z')$	-1.85×10^3	5.21×10^3
$T_o(z')$	7.22×10^4	-4.98

Polynomial Series:

1D Fit:

$$g(z) = -5.86 \times 10^2 + 7.22 \times 10^4 z'$$

$$h(\dot{z}) = -1.33 \times 10^3 + 5.29 \times 10^3 \dot{z}'$$

2D Fit:

$$G(z,\dot{z}) = \sum_{\ell=0}^{m2} \sum_{m=0}^{n2} A_{\ell m} z'^{\ell} \dot{z}'^{m}$$

ℓ \ m	1	$\dot{z}'$
1	-1.28×10^3	40.25
z'	7.22×10^3	-3.77×10^{-3}

FIGURE 19.

Actual System: NONLINEAR (Element 3)

EXACT $G(z,\dot{z})$: $1.49 \times 10^4 z + 60.0\dot{z} - 2.23 \times 10^2 z^3$

Identification Excitation: $S(t) = S_o \sin(at^2 + bt)$

$S_o = 1.0$

Motion Range: $z_{min} = -3.91$ $\quad z_{max} = 3.86$

$\dot{z}_{min} = -1.24 \times 10^2$ $\quad \dot{z}_{max} = 1.24 \times 10^2$

1D Fit: $\hat{G}(z,\dot{z}) = \hat{g}(z) + \hat{h}(\dot{z})$

$$\hat{g}(z) = 6.51 \times 10^3 T_o(z') - 3.37 \times 10^5 T_1(z') + 8.36 \times 10^3 T_2(z') - 1.87 \times 10^5 T_1(z')$$

$$\hat{h}(\dot{z}) = 1.41 \times 10^3 T_o(\dot{z}') - 5.55 \times 10^5 T_1(\dot{z}')$$

Coefficient of 2D Chebyshev series:

$$G(z,\dot{z}) = \sum_{i=0}^{m2} \sum_{j=0}^{n2} C_{ij} T_i(z') T_j(\dot{z}')$$

i \ j	$T_o(\dot{z}')$	$T_1(\dot{z}')$
$T_o(z')$	-9.36×10^2	-4.37×10^3
$T_1(z')$	4.80×10^4	3.30
$T_2(z')$	60.80	-1.29×10^2
$T_3(z')$	-3.18×10^3	8.23×10^{-1}

Polynomial Series:

1D Fit: $g(z) = -4.39 \times 10^2 + 5.76 \times 10^4 z' + 1.7 \times 10^2 z'^2 - 1.28 \times 10^4 z'^3$

$h(\dot{z}) = -6.04 \times 10^2 - 4.48 \times 10^3 \dot{z}'$

2D Fit: $$G(z,\dot{z}) = \sum_{\ell=0}^{m2} \sum_{m=0}^{n2} A_{\ell m} z'^{\ell} \dot{z}'^{m}$$

ℓ \ m	1	$\dot{z}'$
1	-5.88×10^2	-34.24
z'	1.48×10^4	-5.05×10^{-3}
z'^2	-7.97	-1.38×10^{-1}
z'^3	-2.17×10^2	4.54×10^{-4}

FIGURE 20.

Actual System: NONLINEAR (Element 2)

EXACT $G(z,\dot{z})$: $1.49 \times 10^4 z + 60.0\dot{z}$

Identification Excitation: $S(t) = S_o \sin(at^2 + bt)$

$S_o = 1.0$

Motion Range: $z_{min} = -4.87$ $\quad z_{max} = 4.85$

$\dot{z}_{min} = -80.69$ $\quad \dot{z}_{max} = 80.10$

1D Fit:

$$\hat{G}(z,\dot{z}) = g(z) + h(\dot{z})$$

$$\hat{g}(z) = -1.58 \times 10^3 T_o(z') + 3.48 \times 10^5 T_1(z') - 5.25 \times 10^2 T_2(z') - 1.57 \times 10^3 T_3(z')$$

$$\hat{h}(\dot{z}) = -1.63 \times 10^3 T_o(\dot{z}') + 4.33 \times 10^5 T_1(\dot{z}')$$

Coefficient of 2D Chebyshev series:

$$G(z,z) = \sum_{i=0}^{m2} \sum_{j=0}^{n2} C_{ij} T_i(z') T_j(\dot{z}')$$

i \ j	$T_o(\dot{z}')$	$T_1(\dot{z}')$
$T_o(z')$	-3.58×10^2	5.40×10^3
$T_1(z')$	7.25×10^4	-25.59
$T_2(z')$	-24.97	4.15
$T_3(z')$	-62.93	19.40

Polynomial Series:

1D Fit:

$$g(z) = -2.95 \times 10^2 + 7.25 \times 10^4 z' - 46.20 z'^2$$

$$h(\dot{z}) = -46.11 + 5.38 \times 10^3 \dot{z}'$$

2D Fit:

$$G(z,\dot{z}) = \sum_{\ell=0}^{m2} \sum_{m=0}^{n2} A_{\ell m} z'^{\ell} \dot{z}'^{m}$$

ℓ \ m	1	$\dot{z}'$
1	-2.08×10^2	67.11
z'	1.50×10^4	-2.14×10^{-1}
z'^2	-2.17	4.64×10^{-3}
z'^3	-2.19	8.41×10^{-3}

FIGURE 21.

Actual System: NONLINEAR (Element 1)

EXACT $G(z,\dot{z})$: $7214.0z + 21.0\dot{z} + 72.14z^3$

Identification Excitation: $S(t) = S_o \sin(at^2 + bt)$

$S_o = 1.0$

Motion Range: $z_{min} = -8.66$ $\quad z_{max} = 8.61$

$\dot{z}_{min} = -1.39 \times 10^2$ $\quad \dot{z}_{max} = 1.39 \times 10^2$

1D Fit:

$$\hat{G}(z,z) = g(z) + h(\dot{z})$$

$$\hat{g}(z) = 1.23 \times 10^5 T_o(z') + 2.34 \times 10^6 T_1(z') + 1.22 \times 10^5 T_2(z') + 7.64 \times 10^6 T_3(z')$$

$$\hat{h}(\dot{z}) = -5.35 \times 10^2 T_o(\dot{z}') + 4.13 \times 10^5 T_1(\dot{z}')$$

Coefficient of 2D Chebyshev series:

$$G(z,\dot{z}) = \sum_{i=0}^{m2} \sum_{j=0}^{n2} C_{ij} T_i(z') T_j(\dot{z}')$$

i \ j	$T_o(\dot{z}')$	$T_o(\dot{z}')$
$T_o(z')$	-9.31×10^2	2.97×10^3
$T_1(z')$	9.77×10^4	6.27
$T_2(z')$	-1.97×10^2	1.27
$T_3(z')$	1.19×10^2	-11.11

Polynomial series:

1D Fit:

$$g(z) = -2.46 \times 10^2 + 6.22 \times 10^4 z' - 3.56 z'^3 + 4.74 z'^3$$

$$h(\dot{z}) = -5.35 \times 10^2 + 3.00 \times 10^3 \dot{z}'$$

2D Fit:

$$G(z,\dot{z}) = \sum_{\ell=0}^{m2} \sum_{m=0}^{n2} A_{\ell m} z'^{\ell} \dot{z}'^{m}$$

ℓ \ m	1	$\dot{z}'$
1	-1.12×10^3	21.41
z'	7.20×10^3	3.30×10^{-2}
z'^2	-10.39	2.83×10^{-4}
z'^3	73.56	4.97×10^{-4}

FIGURE 22.

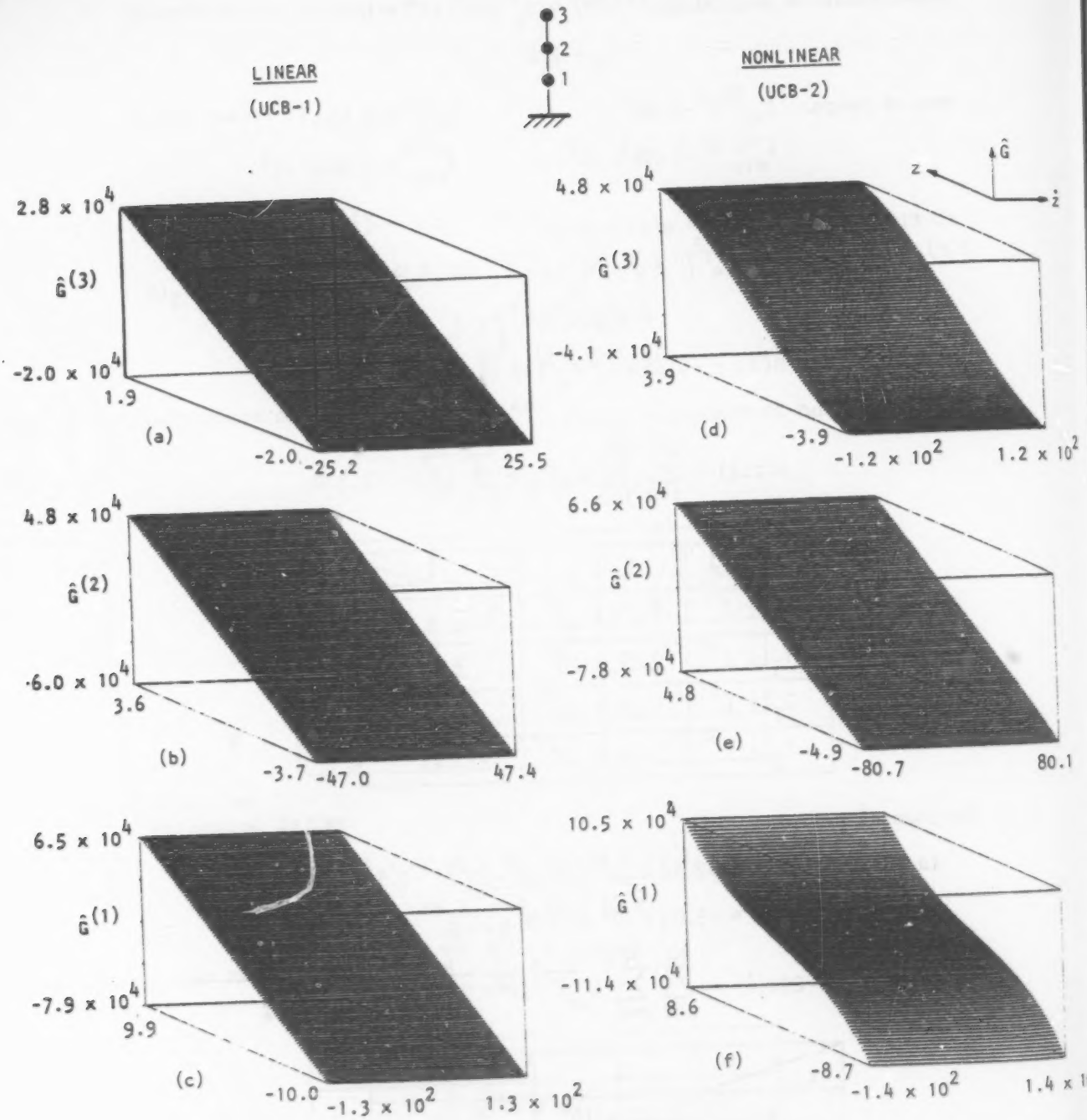


FIGURE 23. RESTORING FUNCTIONS

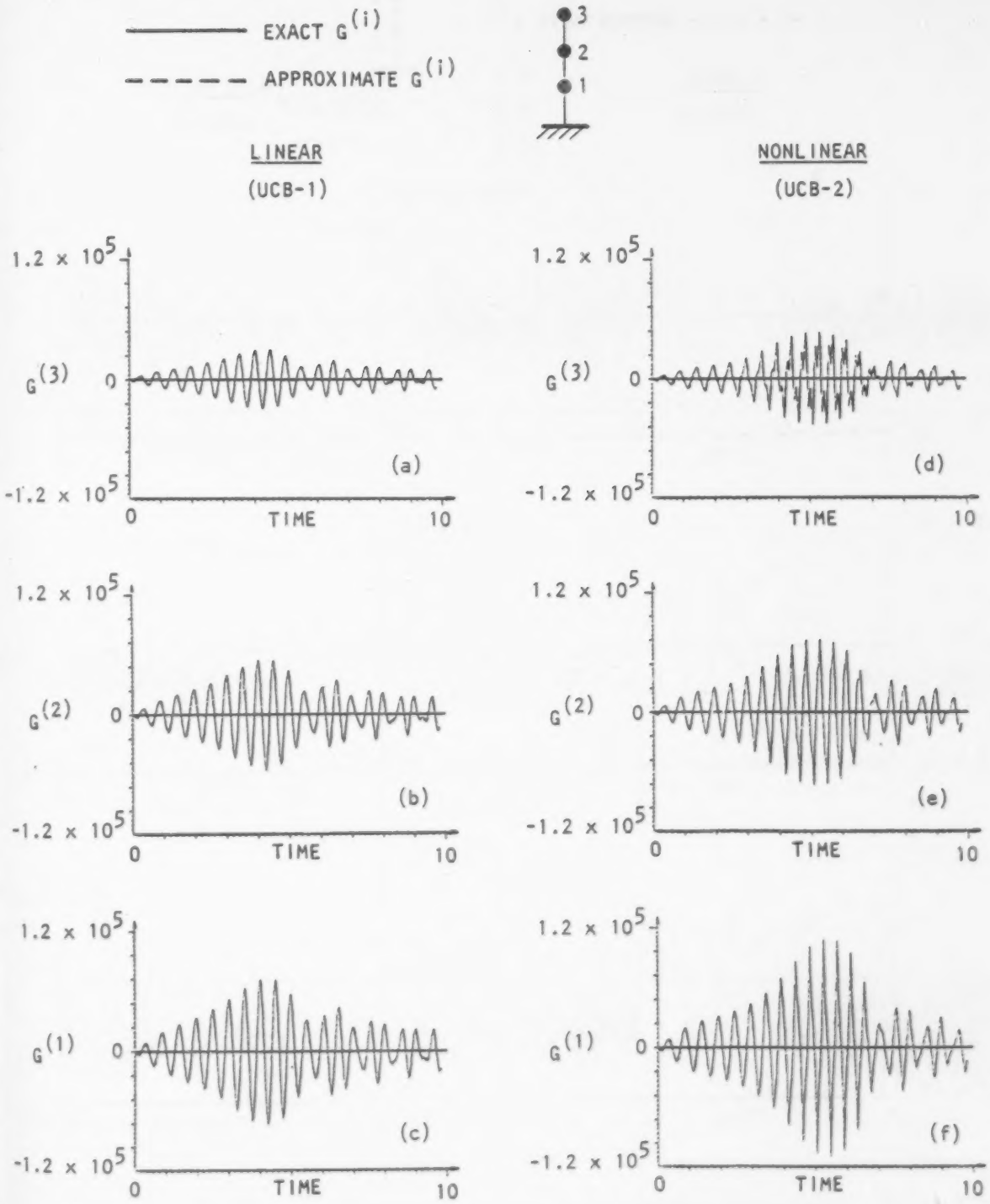


FIGURE 24. COMPARISON BETWEEN EXACT AND APPROXIMATE RESTORING FUNCTIONS

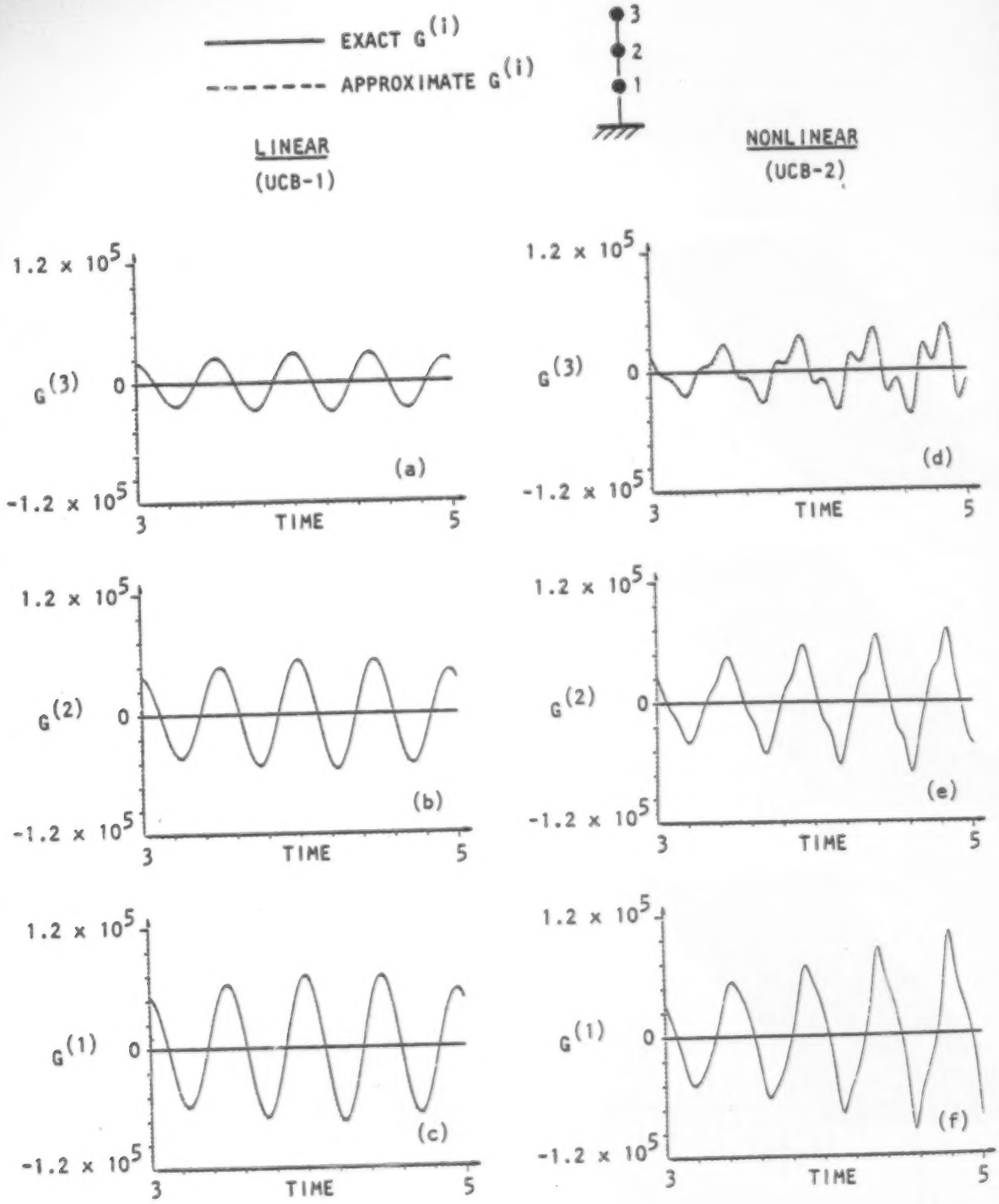


FIGURE 25. COMPARISON BETWEEN EXACT AND APPROXIMATE RESTORING FUNCTIONS

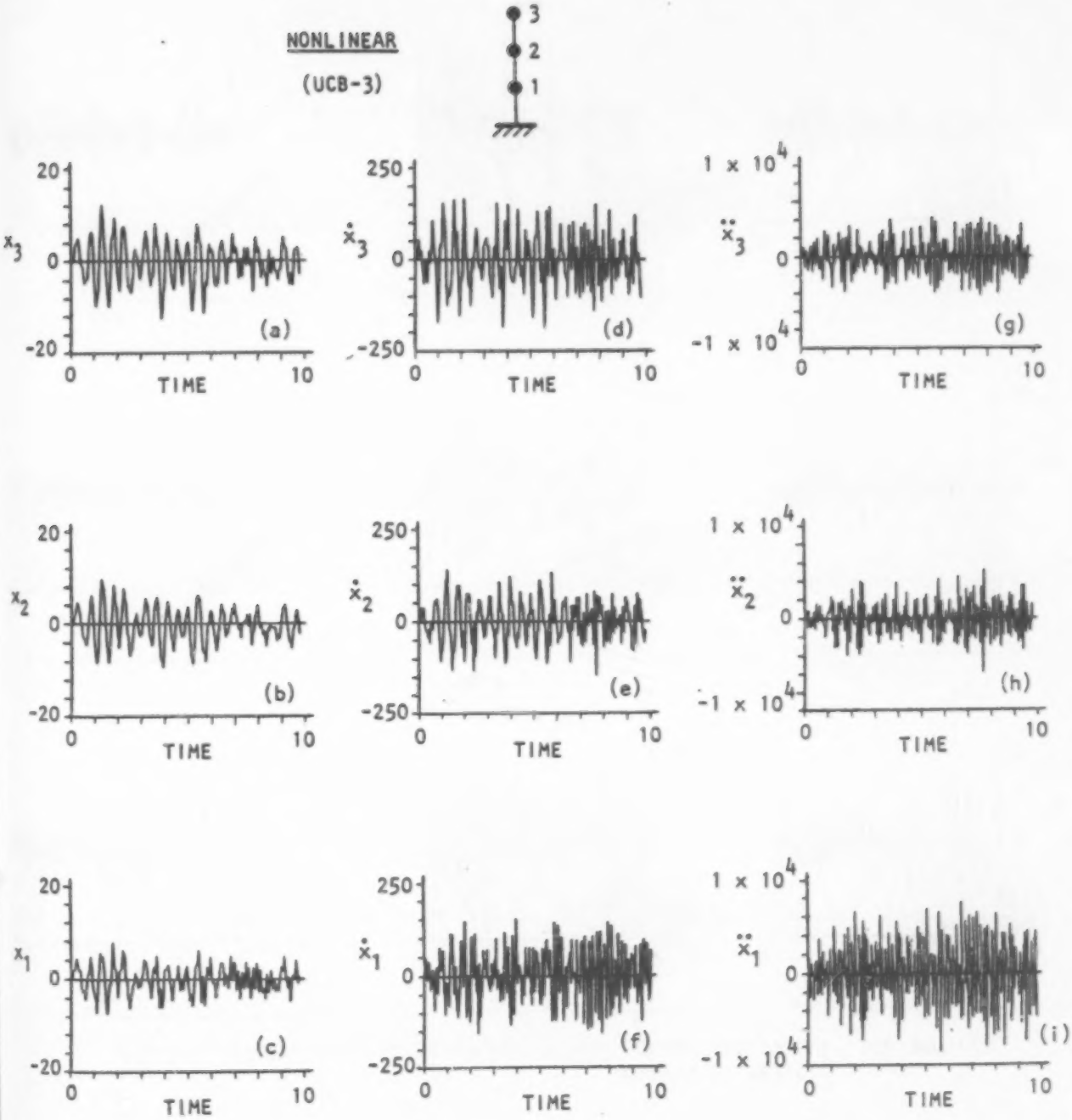


FIGURE 26. RESPONSE TIME HISTORY OF THE HYSTERETIC SYSTEM

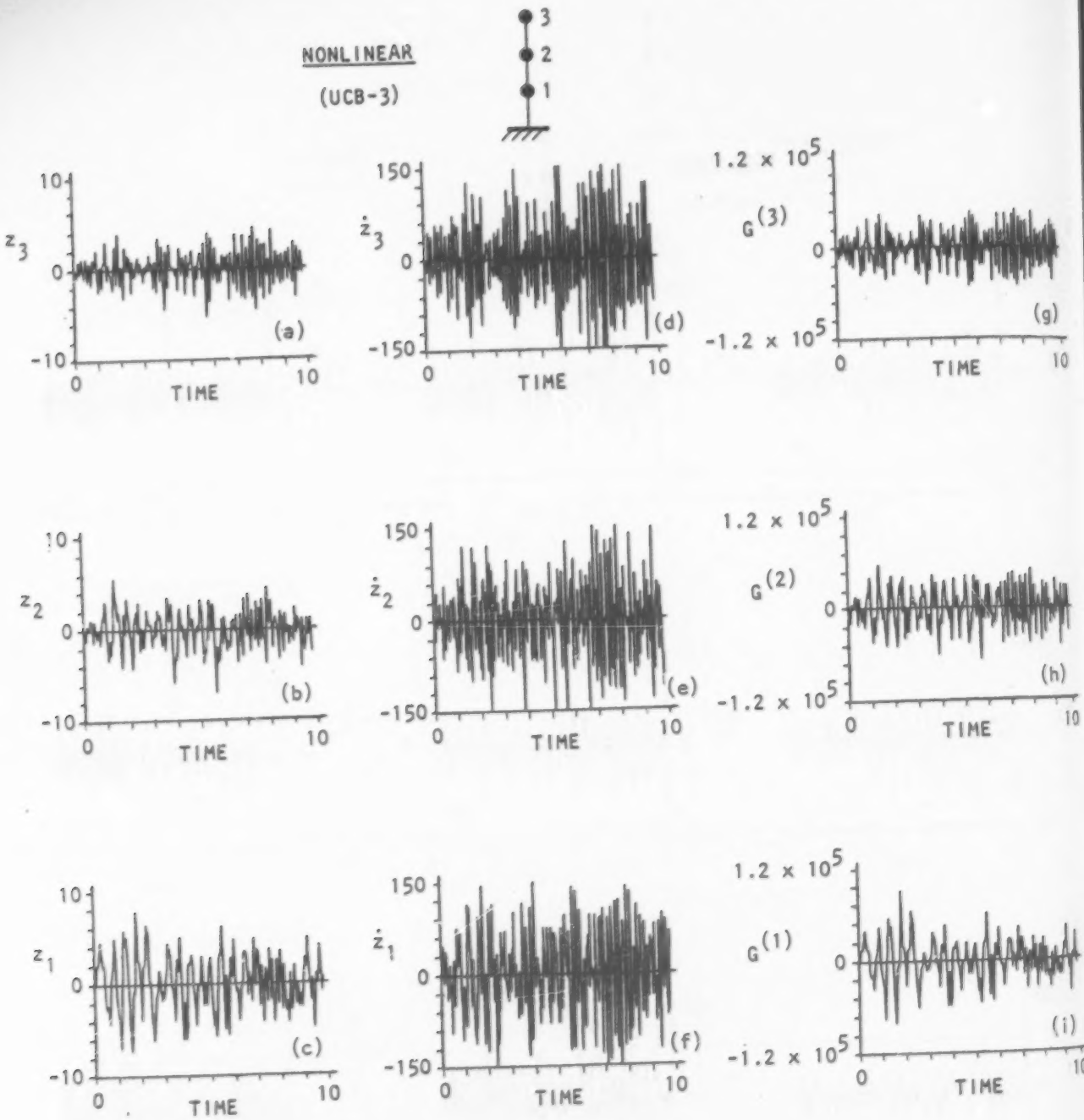


FIGURE 27. INTERSTORY MOTION AND RESTORING FORCE TIME HISTORY FOR THE HYSTERETIC SYSTEM

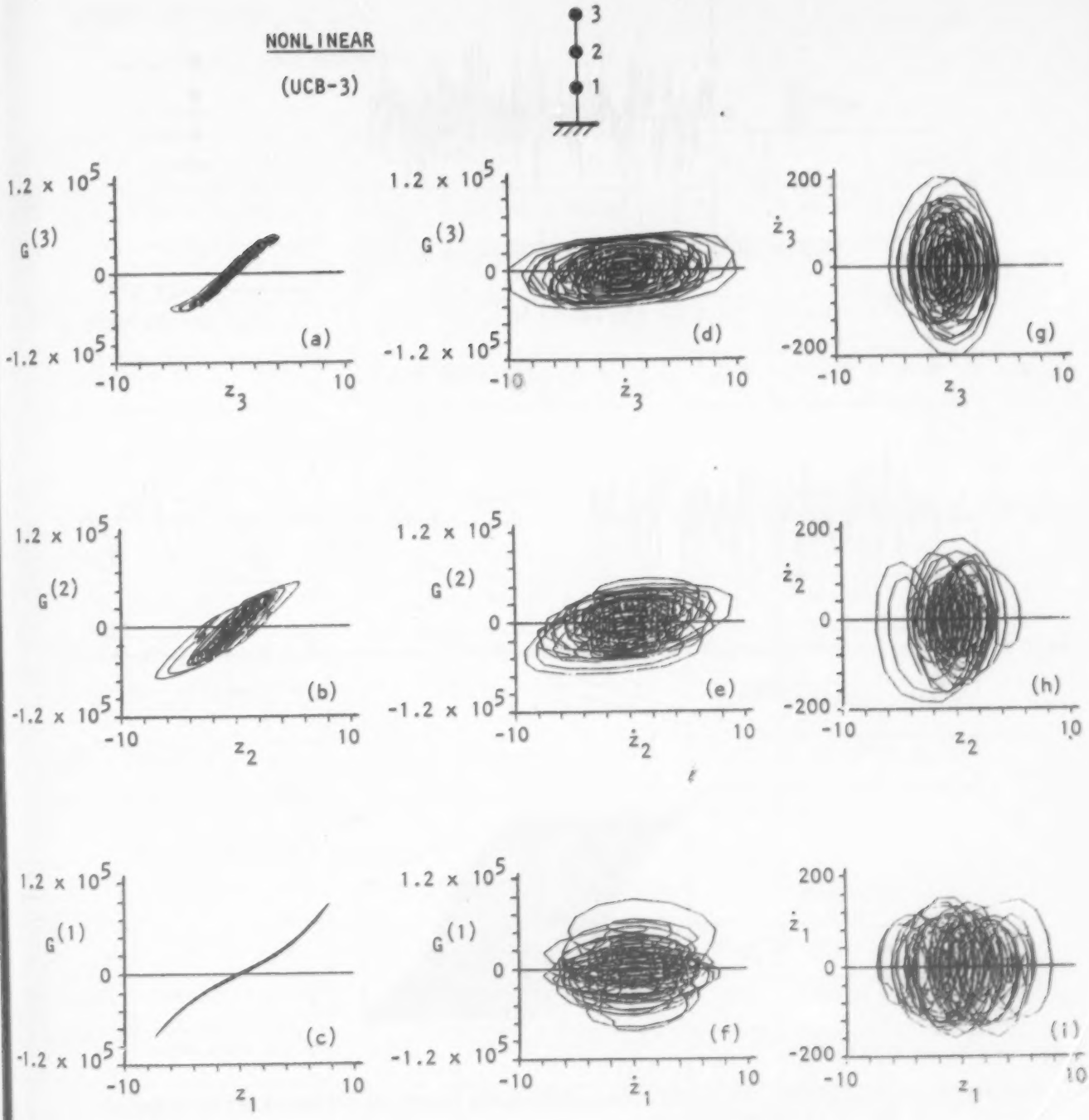


FIGURE 28. RESTORING FUNCTIONS VERSUS THE STATE VARIABLES FOR THE HYSTERETIC SYSTEM

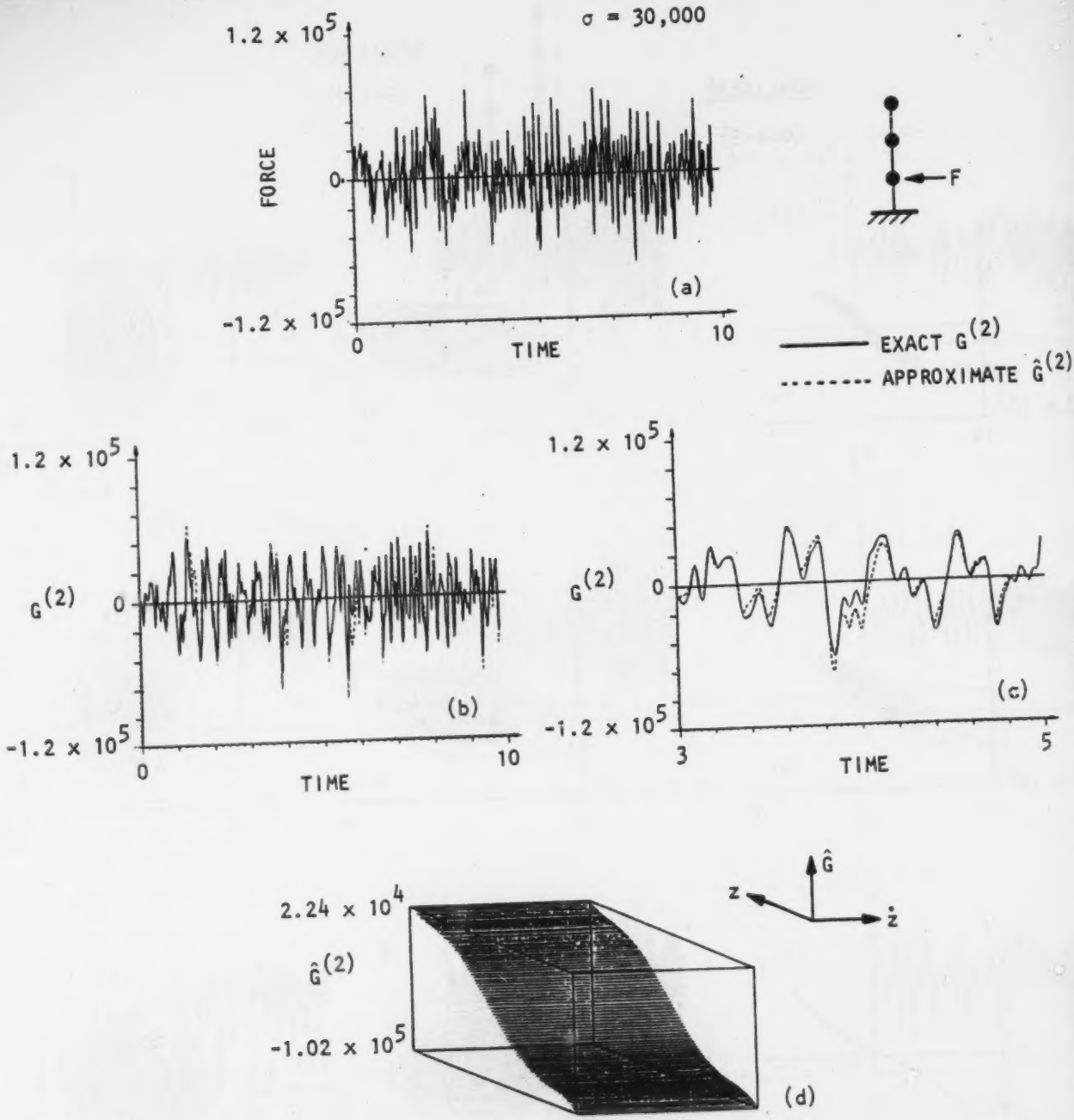


FIGURE 29. EXCITATION, RESPONSE, AND RESTORING FUNCTION OF HYSTERETIC ELEMENT NO. 2 IN MODEL UCB-3

READEX FICHE best copy available

pages tightly bound

2D Fit: $\hat{G}(z,\dot{z}) = \sum_{i=0}^{m2} \sum_{j=0}^{n2} C_{ij}\, T_i(z')\, T_j(\dot{z}')$

Chebyshev Coefficients

Actual System: UCB-2 (Element 3)

No Noise

i \ j	$T_0(\dot{z}')$	$T_1(\dot{z}')$
$T_0(z')$	-1.90×10^3	7.95×10^3
$T_1(z')$	4.44×10^4	-35.99
$T_2(z')$	-2.54×10^2	-2.84×10^2
$T_3(z')$	-2.64×10^3	49.22

1% Noise

i \ j	$T_0(\dot{z}')$	$T_1(\dot{z}')$
$T_0(z')$	-1.88×10^3	7.33×10^3
$T_1(z')$	4.42×10^4	-6.48
$T_2(z')$	-2.23×10^2	-2.76×10^2
$T_3(z')$	-2.30×10^3	32.80

5% Noise

i \ j	$T_0(\dot{z}')$	$T_1(\dot{z}')$
$T_0(z')$	5.46×10^2	7.72×10^3
$T_1(z')$	4.30×10^4	82.45
$T_2(z')$	1.67×10^3	24.47
$T_3(z')$	-3.10×10^3	-94.68

(a)

Actual System: UCB-2 (Element 2)

No Noise

i \ j	$T_0(\dot{z}')$	$T_1(\dot{z}')$
$T_0(z')$	-5.08×10^3	9.34×10^3
$T_1(z')$	5.58×10^4	1.89×10^2
$T_2(z')$	-9.72×10^2	6.00×10^2
$T_3(z')$	2.49×10^2	-3.29×10^2

1% Noise

i \ j	$T_0(\dot{z}')$	$T_1(\dot{z}')$
$T_0(z')$	-4.70×10^3	8.54×10^3
$T_1(z')$	5.59×10^4	1.57×10^2
$T_2(z')$	-8.39×10^2	3.91×10^2
$T_3(z')$	2.30×10^2	-2.84×10^2

5% Noise

i \ j	$T_0(\dot{z}')$	$T_1(\dot{z}')$
$T_0(z')$	-4.94×10^3	5.56×10^3
$T_1(z')$	5.42×10^4	95.05
$T_2(z')$	-3.88×10^2	-5.91×10^2
$T_3(z')$	-1.13×10^3	-3.06×10^2

(b)

Actual System: UCB-2 (Element 1)

No Noise

i \ j	$T_0(\dot{z}')$	$T_1(\dot{z}')$
$T_0(z')$	7.88×10^3	4.10×10^3
$T_1(z')$	6.95×10^4	-2.59×10^2
$T_2(z')$	3.03×10^3	-3.03×10^2
$T_3(z')$	5.65×10^3	2.28×10^2

1% Noise

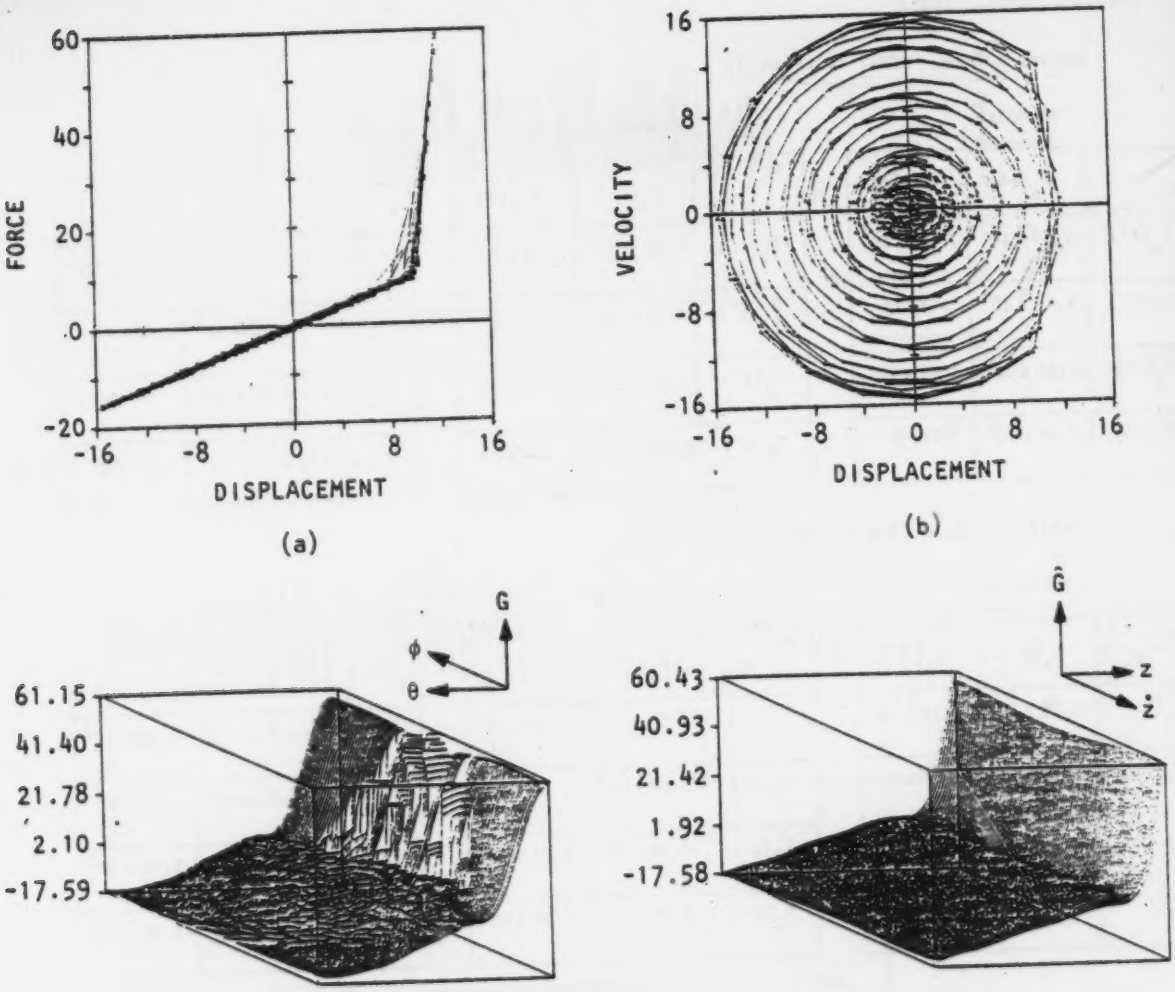
i \ j	$T_0(\dot{z}')$	$T_1(\dot{z}')$
$T_0(z')$	7.67×10^2	3.43×10^3
$T_1(z')$	6.90×10^4	-3.63×10^2
$T_2(z')$	3.37×10^3	-1.49×10^2
$T_3(z')$	6.19×10^3	4.76×10^2

5% Noise

i \ j	$T_0(\dot{z}')$	$T_1(\dot{z}')$
$T_0(z')$	8.16×10^3	3.99×10^3
$T_1(z')$	6.95×10^4	-7.35×10^2
$T_2(z')$	4.24×10^3	-6.07×10^2
$T_3(z')$	5.65×10^3	1.27×10^3

(c)

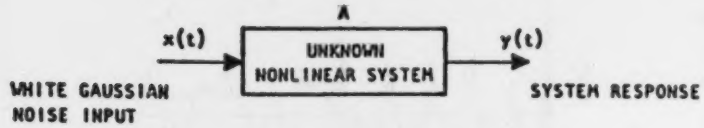
FIGURE 30. CHEBYSHEV COEFFICIENTS FOR UCB-2



(c) Experimental data

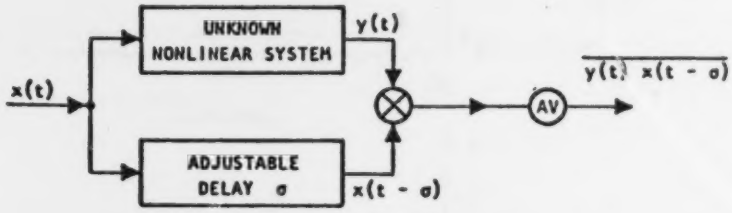
(d) Chebyshev approximation

FIGURE 31. CHARACTERISTICS OF A NONLINEAR ELEMENT WITH A GAP

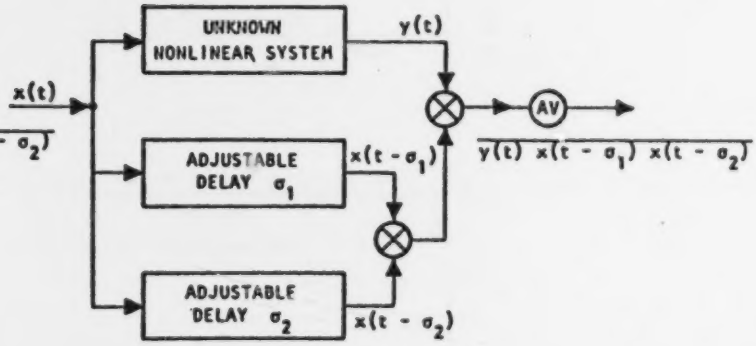


$g_0 = \overline{y(t)}$ = AVERAGE VALUE OF OUTPUT $y(t)$ FOR INPUT $x(t)$

$$g_1(\sigma) = \frac{1}{K}\,\overline{y(t)\,x(t-\sigma)}$$



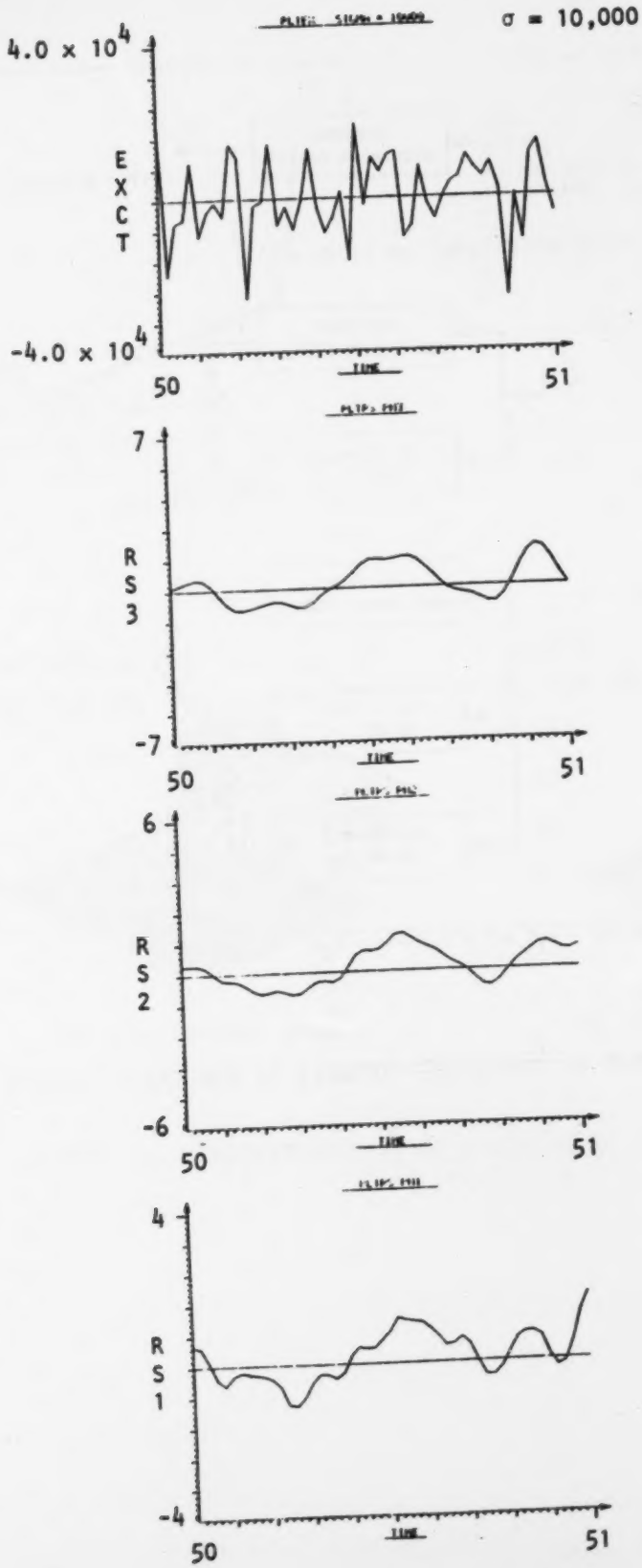
$$g_2(\sigma_1, \sigma_2) = \frac{1}{2K^2}\,\overline{y(t)\,x(t-\sigma_1)\,x(t-\sigma_2)}$$

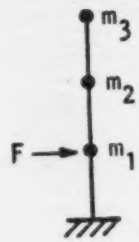


$$g_3(\sigma_1, \sigma_2, \sigma_3) = \frac{1}{6K^3}\,\overline{y(t)\,x(t-\sigma_1)\,x(t-\sigma_2)\,x(t-\sigma_3)}$$

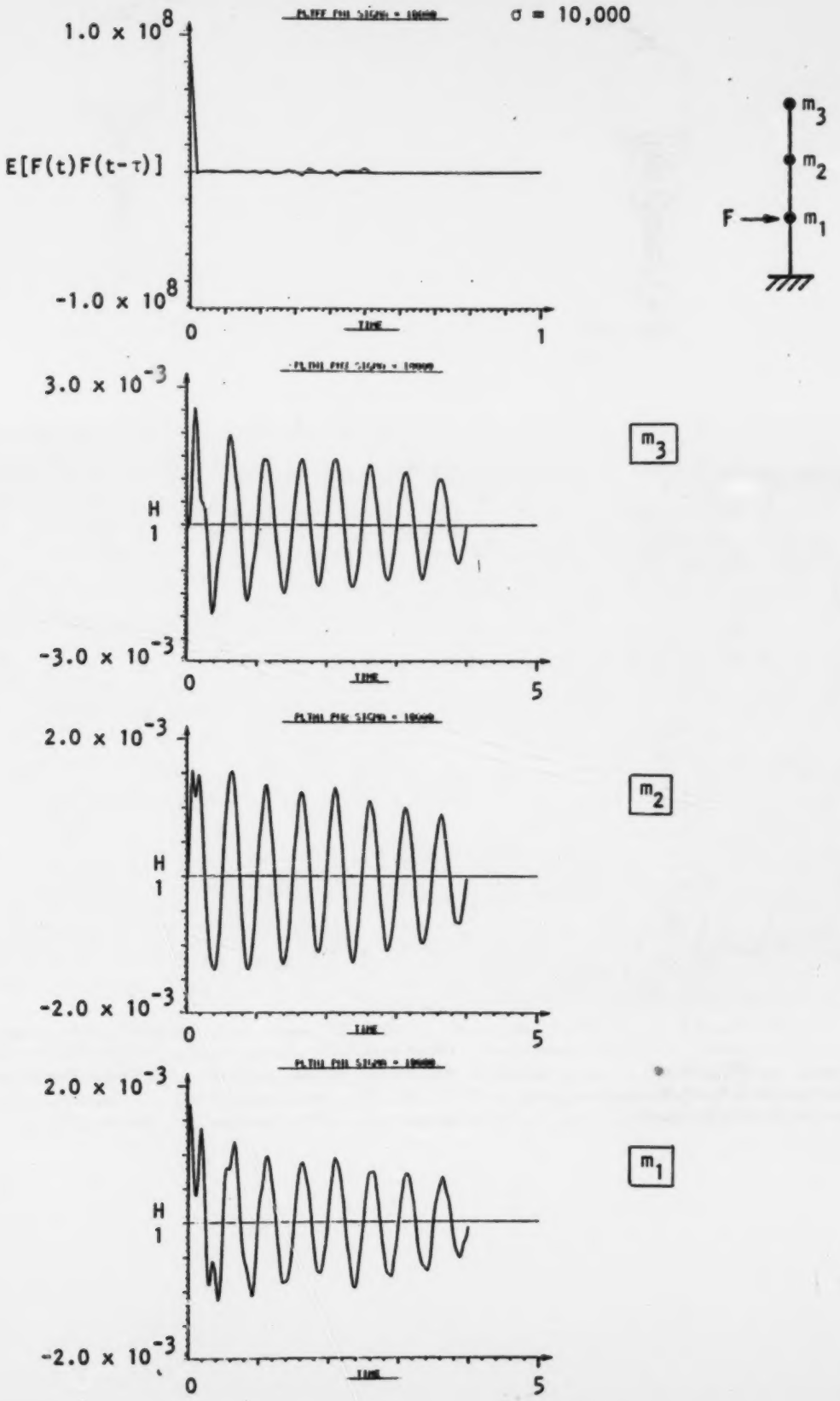
FIGURE 1. MEASUREMENT OF THE WIENER KERNELS OF NONLINEAR SYSTEMS

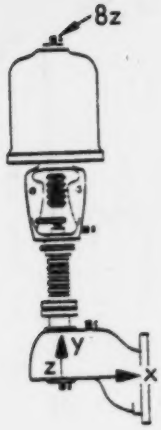
WIENER KERNEL DATA

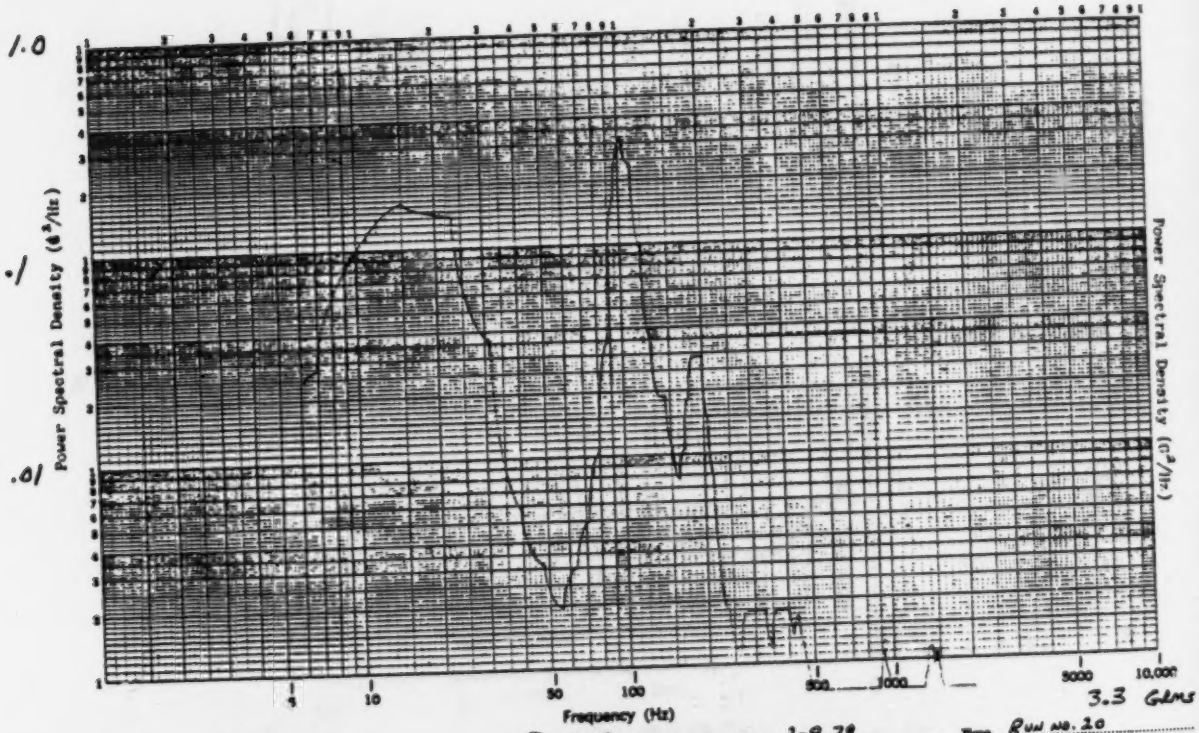




WIENER KERNELS OF A NONLINEAR SYSTEM



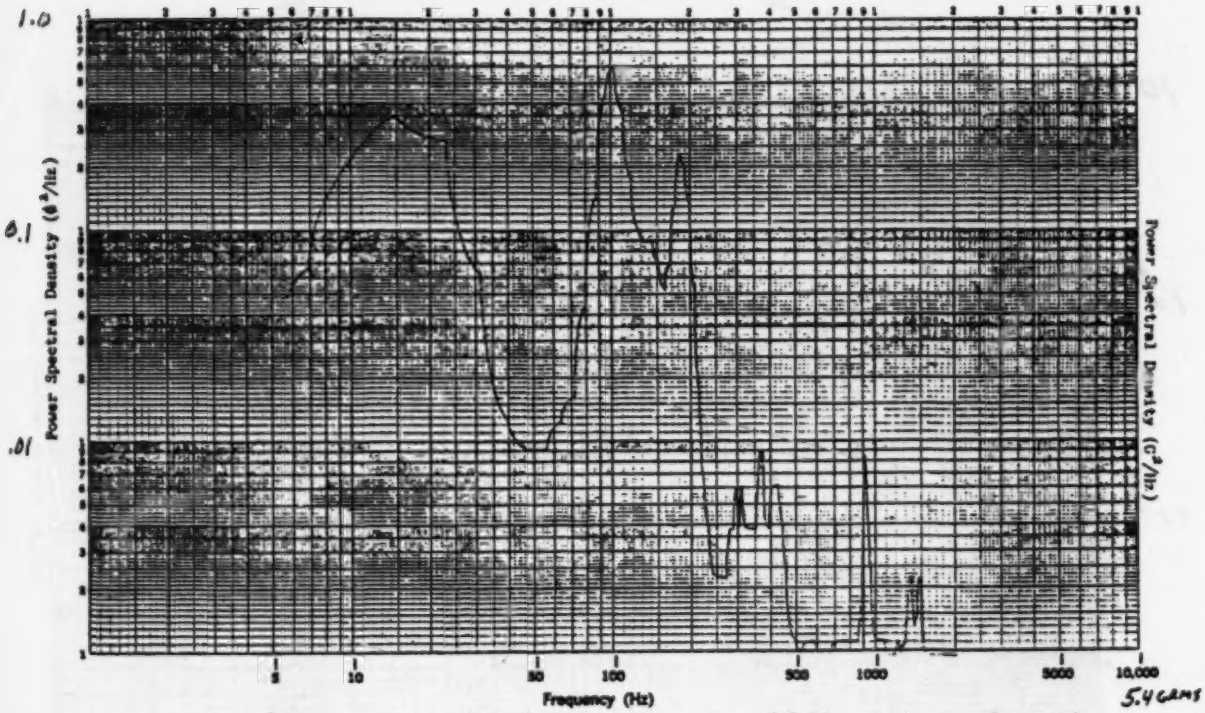




Project No. USE Item NRC VALVE Serial No. Z AXIS Date 2-9-78 Time Run No. 20

Conditions RANDOM VIB. @ % .001g²/Hz Pickup S/N & Location ACCEL. No. 8 Pickup Sensitivity 100mv/g Analyzer B 13 (Hz)

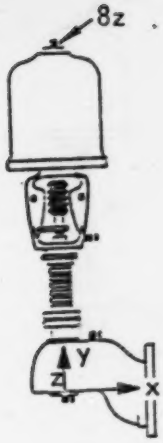
Recorder/Reproducer AMPEX/HONEYWELL Reproduce tape speed 7.5 ips FM/direct Track No. 8

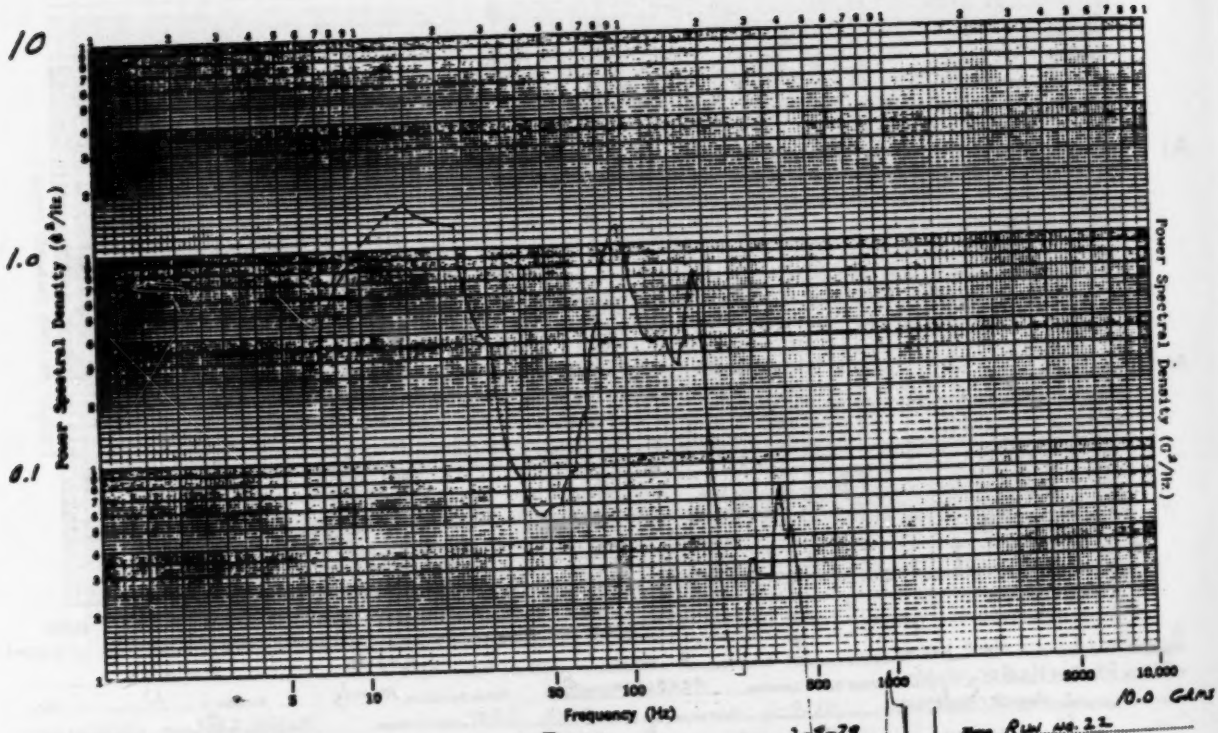


Project No. USC Item NRC VALVE Satellite Z AXIS Date 2-9-78 Time Run no. 21

Conditions RANDOM VIB @ .01 g²/Hz Pickup S/N & Location ACCEL. No. 8 Pickup Sensitivity 100mv/g Analyzer # 13 (Hz)

Recorder/Reproducer Ampex/Honeywell Reproduce tape speed 7.5 ips FM/direct Track No. 8

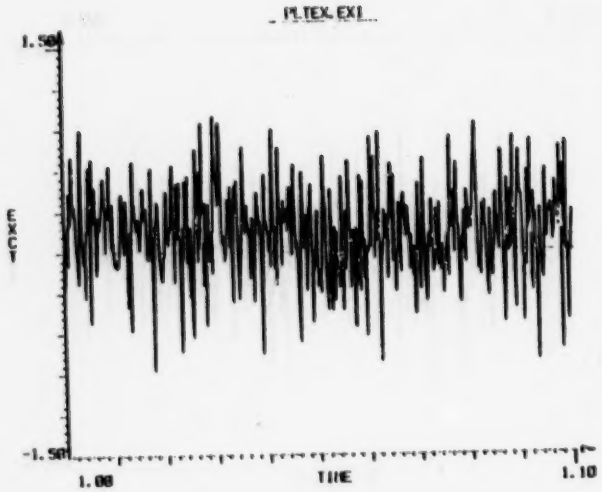


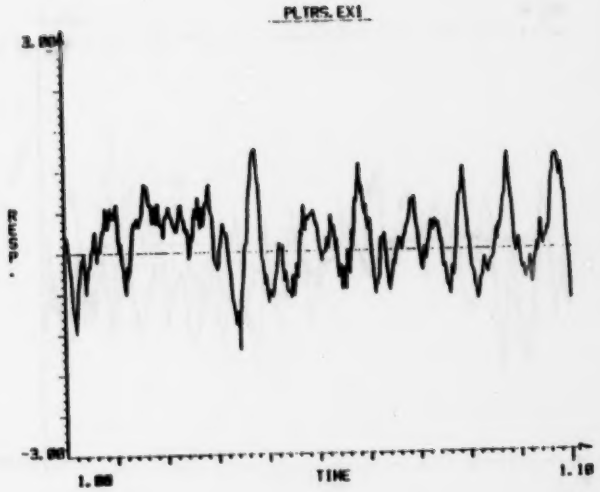


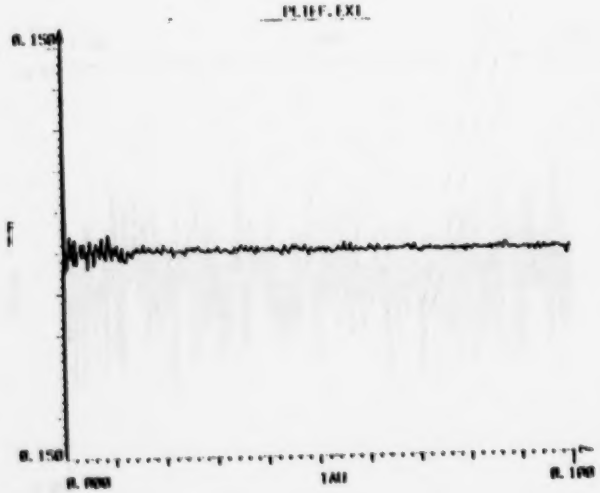
Project No. USC Item NRC VALVE Switch No. Z AXIS Date 2-9-78 Time RUN NO. 22

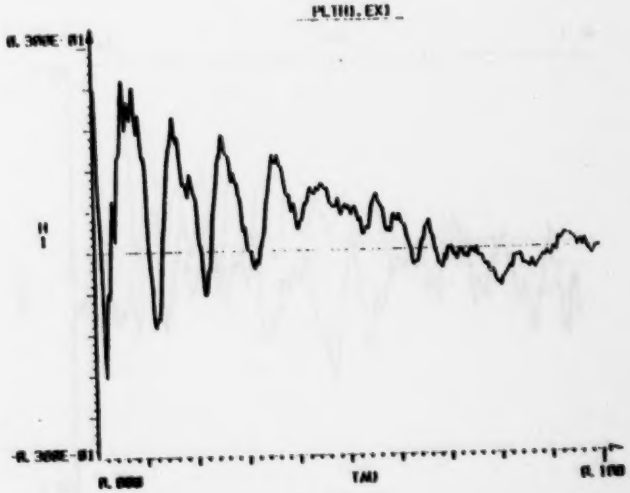
Conditions RANDOM VIB @ ~.02g²/HZ Pickup S/N & Location ACCEL. NO. 8 Pickup Sensitivity 100mv/g Analyzer # 13 (Hz)

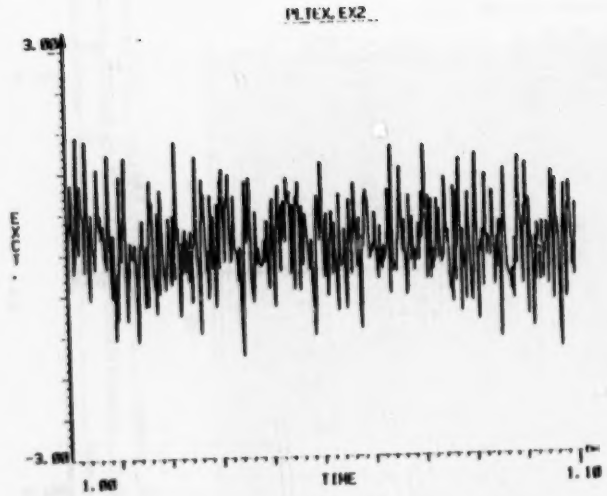
Recorder/Reproducer AMPEX/HONEYWELL Reproduce tape speed 7.5 ips FM/direct Track No. 8

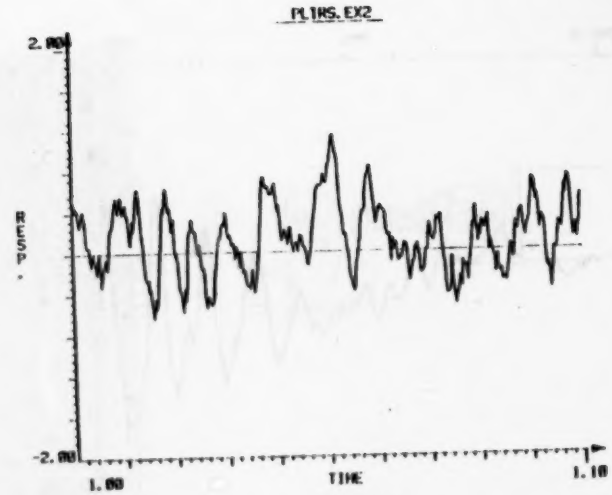

PLTEX.EXI
1.50
-1.50
EXC
1.00
TIME
1.10

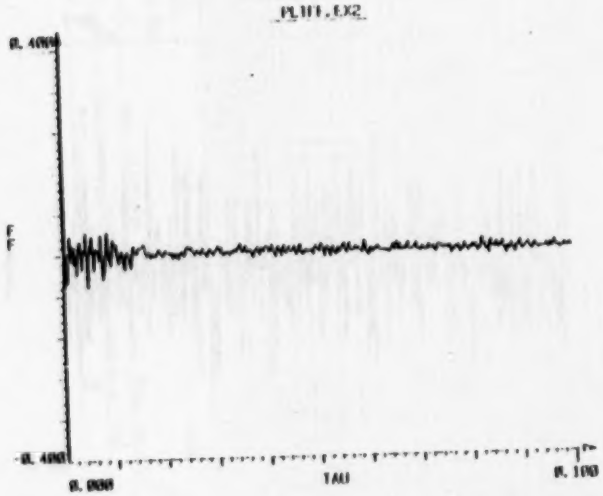

PLTRS.EXI
3.00
-3.00
RESPS
1.00
TIME
1.10

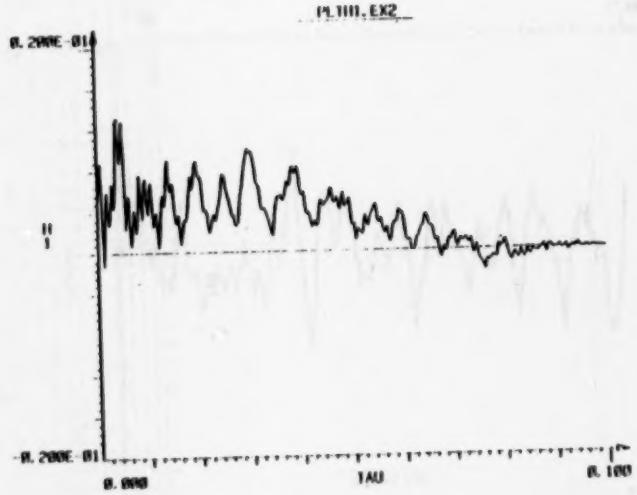

PLTEF.EXI
0.150
-0.150
F
0.000
TAU
0.100

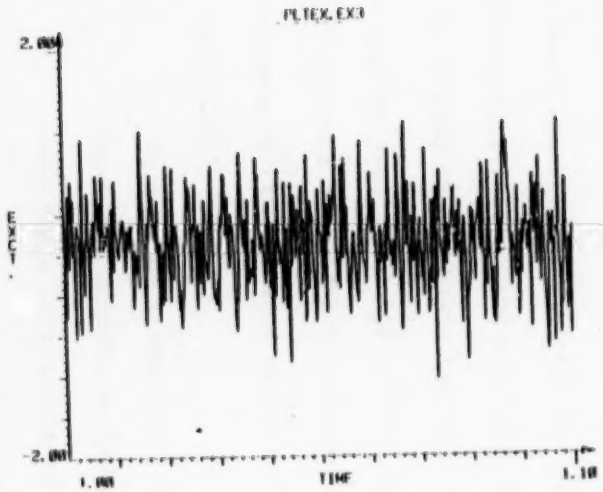

PLTHI.EXI
0.300E-01
-0.300E-01
H
0.000
TAU
0.100

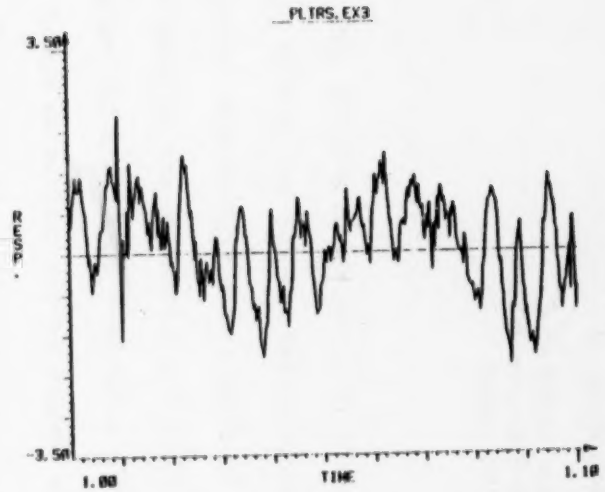

PLTEX.EX2
3.00
EXCT.
-3.00
1.00
TIME
1.10

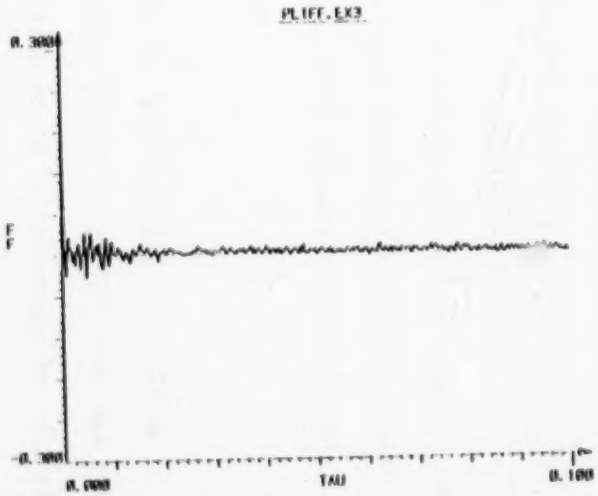

PLTRS.EX2
2.00
RESP.
-2.00
1.00
TIME
1.10

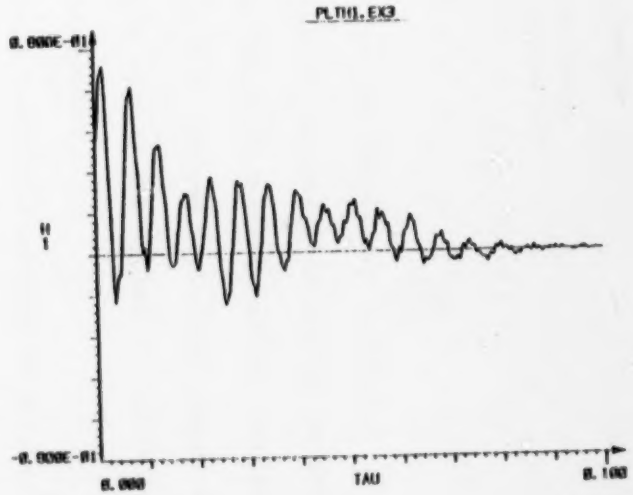

PLTFT.EX2
0.400
F
F
-0.400
0.000
TAU
0.100


PLTHH.EX2
0.200E-01
H
H
-0.200E-01
0.000
TAU
0.100


PLTEX.EX3
2.00
-2.00
EXCT.
1.00
TIME
1.10


PLTRS.EX3
3.50
-3.50
RESP.
1.00
TIME
1.10


PLTFF.EX3
0.300
-0.300
FF
0.000
TAU
0.100


PLTHI.EX3
0.000E-01
-0.000E-01
HI
0.000
TAU
0.100

Mechanical Piping Benchmark Problems

M. Reich and P. Bezler

Structural Analysis Group
Brookhaven National Laboratory
Upton, NY 11973

This program is supported by the U.S. Nuclear Regulatory Commission (NRC), Division of Reactor Safety Research (RSR) and provides support to the Mechanical Engineering Branch (MEB) of the Office of Nuclear Reactor Regulation (NRR) in the evaluation and verification of various structural computer programs and analytical procedures presently used in the nuclear industry for the design of nuclear piping systems and components.

The objective of this work is to develop analytical as well as physical model-test solutions which will be used as specified benchmark problems for piping structures subjected to deadweight, internal pressure, and dynamic loads due to earthquake motion. The piping structures may deform elastically or elastic-plastically, and may contain gapped supports or snubbers.

In FY 1980 the emphasis has been on the verification of the computer programs used by applicants for nuclear power plant licenses to perform linear elastic analyses of piping structures subjected to seismic excitation using the response spectrum method of evaluation.

Specifically:

1) To provide a consistent basis for verification of programs used for seismic analysis, seven piping benchmark problems and solutions were developed. These could be incorporated into the licensing procedure by requiring applicants to provide corresponding solutions. A report, NUREG CR-1677, describing these benchmarks has been issued.

2) Responding to the need for immediate verification, confirmatory evaluations were performed for twelve nuclear power plants, involving seven computer programs. Deficiencies were uncovered in the analysis method/applicant implementation for three of these applicants.

3) To facilitate the confirmatory evaluations and benchmark problem developments, the elastic analysis code EPIPE was updated and a revised user's manual issued. This code was made available to NRC designated contractors and is being used for licensing evaluation.

4) To allow verification of applicant compliance with ASME and nuclear standards the development of two computer codes PSAFE1 and PSAFE2 were completed. PSAFE1 is a post processor designed to perform stress evaluations in accordance with ASME-BP Section III, Subarticle NB-3600, 1974 Ed. PSAFE2 is an elastic analysis code with the capability to perform stress evaluations of Class 2,3 piping per ASME Section III NC/ND. Both codes were made available to NRC designated contractors and is being used for licensing evaluation.

MECHANICAL PIPING BENCHMARK PROBLEMS

- DYNAMIC STRUCTURAL ANALYSIS OF PIPING SYSTEMS IS ONE OF THE MOST EXTENSIVE ENGINEERING EFFORTS REQUIRED FOR THE SAFETY DESIGN OF NUCLEAR POWER PLANTS.
- SUCH ANALYSIS IS NORMALLY PERFORMED USING COMPUTER PROGRAMS WHICH CAN HANDLE COMPLEX GEOMETRIES AND VARIOUS STATIC AND DYNAMIC LOADING CONDITIONS.
- APPLICANTS FOR NUCLEAR POWER PLANT LICENSES ARE REQUIRED TO PROVIDE CONFIRMATION OF THE ADEQUACY OF THEIR PROGRAMS, AS PRESCRIBED BY THE GUIDELINES OF THE STANDARD REVIEW PLAN (NUREG-75/087, SEPT. 1975) AND APPENDIX B, SECTION III, OF 10CFR50.

- OF PARTICULAR CONCERN IS THE CLASS OF PROGRAMS WHICH ARE USED BY THE INDUSTRY TO PERFORM DYNAMIC STRUCTURAL ANALYSES OF COMPLEX PIPING SYSTEMS USING THE RESPONSE SPECTRUM METHOD.

- THESE ARE GENERALLY LARGE PROGRAMS, BASED ON THE FINITE ELEMENT METHOD, WHICH CONSIDER THE STRUCTURES TO BE ELASTIC OVER THE ENTIRE DEFORMATION HISTORY AND TO EXPERIENCE SMALL DISPLACEMENTS AND ROTATIONS.

- SOME OF THESE PROGRAMS ARE COMMONLY AVAILABLE GENERAL PURPOSE PROGRAMS, WHILE OTHERS ARE OF A PROPRIETARY NATURE.

A NEED HAS BEEN DETERMINED FOR THE ADDITIONAL VERIFICATION OF THE COMPUTER PROGRAMS USED TO PERFORM DYNAMIC STRUCTURAL ANALYSIS OF COMPLEX PIPING SYSTEMS BEYOND THE LEVEL CURRENTLY PROVIDED BY APPLICANTS FOR NUCLEAR POWER PLANT LICENSES.

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

OBJECTIVE:

ASSIST MEB IN THE EVALUATION AND VERIFICATION OF THE VARIOUS COMPUTER PROGRAMS AND ANALYTICAL PROCEDURES USED IN THE NUCLEAR INDUSTRY FOR THE DESIGN OF NUCLEAR PIPING SYSTEMS AND COMPONENTS.

FY 80 PROGRAM TASKS

1) EVALUATION OF APPLICANT SOLUTIONS

2) ELASTIC BENCHMARK DEVELOPMENT

3) CLASS 1 AND CLASS 2, 3 PIPING ANALYSIS CODES

4) ELASTIC PIPING ANALYSIS CODE OPTIMIZATION

5) EVALUATION OF APPLICANT ANALYSIS METHODS

FY 81/82 PROGRAM TASKS

6) RESPONSE OF PIPING TO TIME DEPENDENT LOADINGS

7) INELASTIC ANALYSIS CODE MODIFICATION AND BENCHMARK DEVELOPMENT

8) PHYSICAL BENCHMARK TESTS

EVALUATION OF APPLICANT SOLUTIONS

(CONFIRMATORY EVALUATIONS)

USING THE VERIFIED, BNL DEVELOPED ELASTIC ANALYSIS CODE EPIPE, RESPONSE SPECTRUM SOLUTIONS WERE DEVELOPED FOR SELECTED PIPING PROBLEMS FOR SYSTEMS FROM EXISTING PLANTS. COMPARISON OF THESE SOLUTIONS TO THE SOLUTIONS DEVELOPED BY THE APPLICANT PERMIT THE VERIFICATION OF THE APPLICANTS ANALYSIS METHOD AND ASME AND NUCLEAR STANDARDS COMPLIANCE.

EPIPE CODE MODIFICATIONS FOR CONFIRMATORY ANALYSES

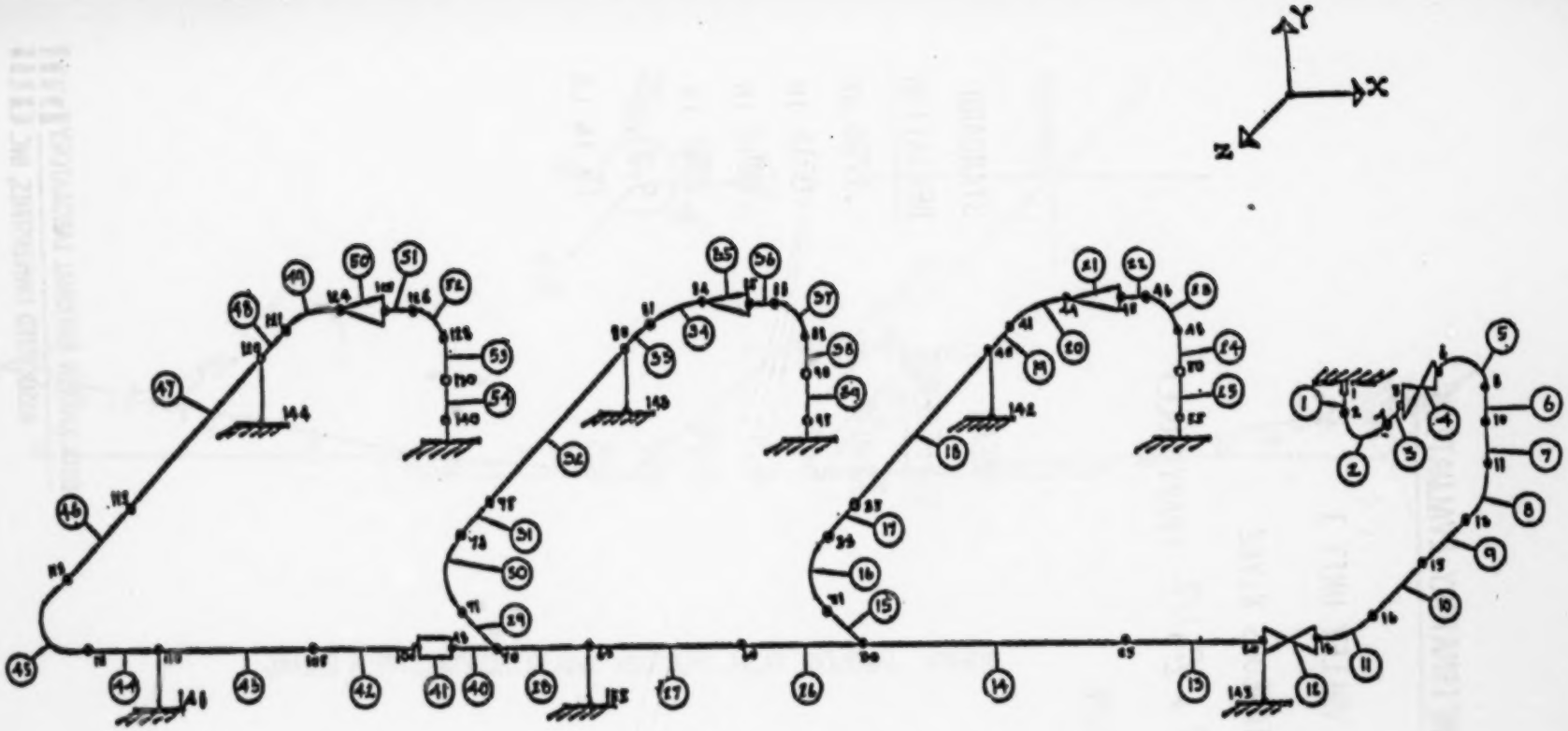
- INCORPORATED ALGEBRAIC, ABSOLUTE SUM, NAVY RULE AND SRSS SPATIAL COMBINATION RULE ALGORITHMS
- INCORPORATED MULTI-DIRECTIONAL EXCITATION CAPABILITY
- INSTALLED LEVEL 2 UPDATE
- INCORPORATED BANDWIDTH MINIMIZATION SCHEME

CONFIRMATORY EVALUATIONS COMPLETED

PLANT	APPLICANT COMPUTER CODE	APPLICANT	COMMENTS
MAINE YANKEE	NUPIPE,SHOCK3	S&W	NAVY RULE COMB
SURRY 1	NUPIPE	S&W	
BEAVER VALLEY 1	NUPIPE	S&W	NAVY RULE COMB
JAMES A. FITZPATRICK	SHOCK3	S&W	
BRUNSWICK	P1SYS02	GE	
BRUNSWICK	ADLPIPE 2	UE&C	MODELING ERRORS
INDIAN PT. 3	ADLPIPE 2	UE&C	MODELING ERRORS
NINE MILE PT. 1	ADLPIPE	TELEDYNE	ALG COMB
NINE MILE PT. 1	TMRSAP	TELEDYNE	
PILGRIM 1	STARDYNE 3	TELEDYNE	
SURRY 2	NUPIPE	EBASCO	
DONALD C. COOK 1	WESTDYN	WESTINGHOUSE	
SALEM 1	PIPDYN 2	PSE&G	POOR AGREEMENT
BRUNSWICK	ADLPIPE 2	UE&C	POOR AGREEMENT
SALEM 2	PIPDYN 2	PSE&G	

CONFIRMATORY EVALUATION RESULTS

CONFIRMATORY EVALUATIONS HAVE UNCOVERED DEFICIENCIES IN THE ANALYSIS METHODS/APPLICANT IMPLEMENTATION OF THREE APPLICANT LICENSES. THEY HAVE CONFIRMED THE ADEQUACY OF THE ANALYSIS METHODS OF FOUR APPLICANT LICENSES.



BEAVER VALLEY UNIT 1 MAIN STEAM LINE (NUPIPE)

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

CONFIRMATORY EVALUATION

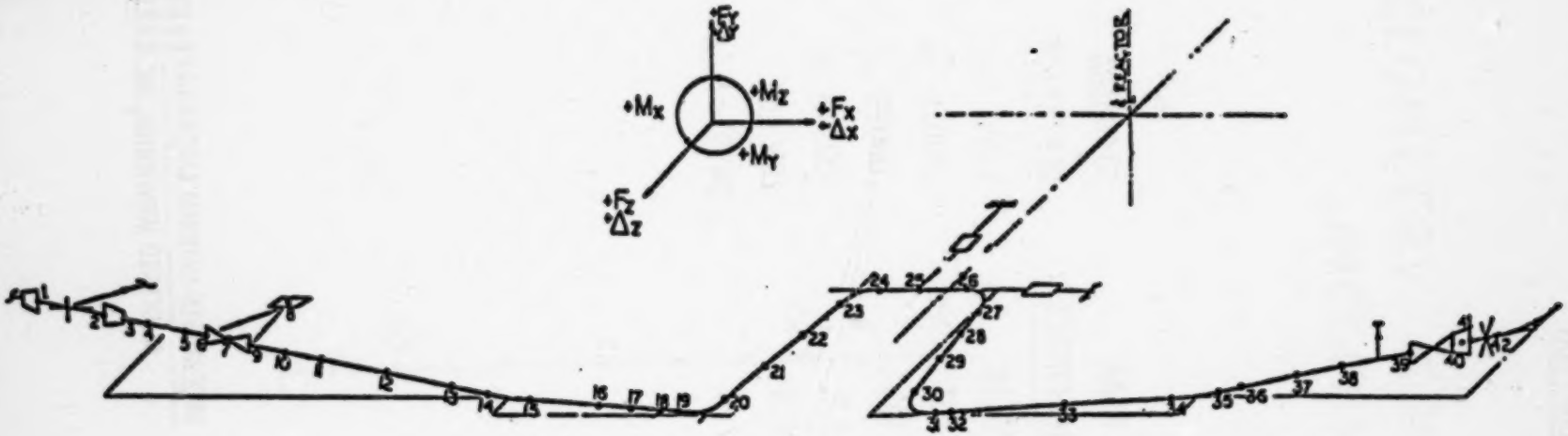
NUCLEAR PLANT: BEAVER VALLEY UNIT 1

EXCITATION: OBE SIMULTANEOUS X,Y&Z

COMBINATION: $[Y] + (X^2 + Z^2)^{1/2}$ (NAVY RULE)

RESULTS: NUPIPE VS. EPIPE

	MEAN DIFFERENCE	STANDARD DEVIATION
FREQUENCY	2.0 %	.1796 HZ
X DISPLACEMENT	6.2	.0018 IN
Y DISPLACEMENT	6.5	.0048 IN
Z DISPLACEMENT	5.75	.0097 IN
RESULTANT FORCE	6.7	5.2 LB
RESULTANT MOMENT	4.5	15 IN LB



JAMES A. FITZPATRICK PLANT RESIDUAL HEAT REMOVAL PIPING

CONFIRMATORY EVALUATION

NUCLEAR PLANT: JAMES A. FITZPATRICK

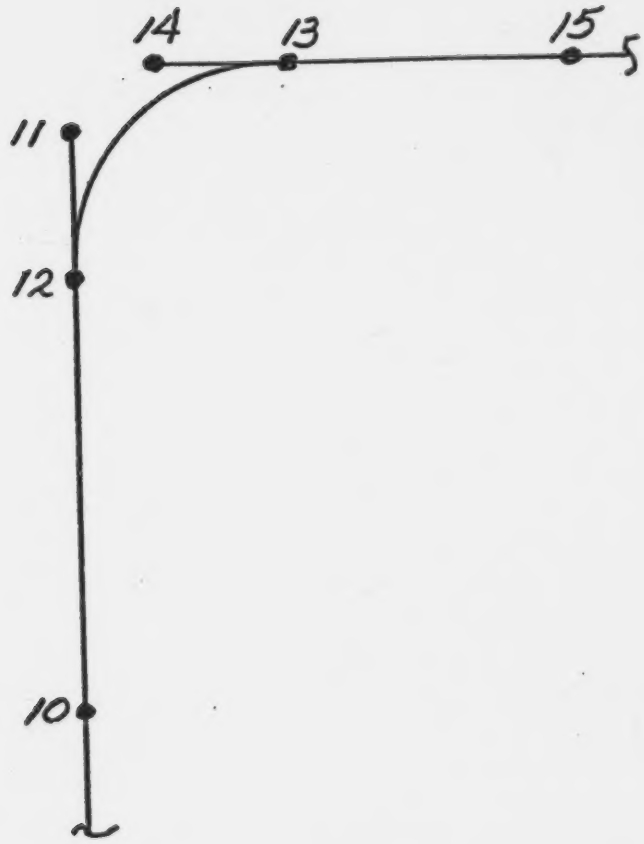
EXCITATION: OBE SIMULTANEOUS X,Y&Z

COMBINATION: SRSS

RESULTS: SHOCK3 VS. EPIPE

	MEAN DIFFERENCE	STANDARD DEVIATION
FREQUENCY	3.1%	1.047 HZ
X DISPLACEMENT	1.3	.00029 IN
Y DISPLACEMENT	38.	.00098 IN
Z DISPLACEMENT	3.6	.00148 IN
RESULTANT FORCE	5.9	186 LB
RESULTANT MOMENT	7.8	2439 FT LB

GEOMETRY ERROR OF TYPE UNCOVERED



BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

ELASTIC BENCHMARK DEVELOPMENT

THE QUALITY OF APPLICANT PIPING ANALYSIS METHODS AND COMPUTER PROGRAMS MAY BE DETERMINED FROM A COMPARISON OF APPLICANT GENERATED SOLUTIONS WITH KNOW SOLUTIONS OF SELECTED BENCHMARK PROBLEMS. A SET OF SUCH PROBLEMS AND CORRESPONDING SOLUTIONS WERE DEVELOPED.

ACCOMPLISHMENTS

- SEVEN BENCHMARK PROBLEMS AND SOLUTIONS FOR SEISMIC ANALYSIS WERE DEVELOPED USING MULTIDIRECTIONAL EXCITATION AND THE RESPONSE SPECTRUM ANALYSIS METHOD. THESE COULD BE INCORPORATED INTO THE LICENSING PROCEDURE BY REQUIRING APPLICANTS TO PROVIDE CORRESPONDING SOLUTIONS.

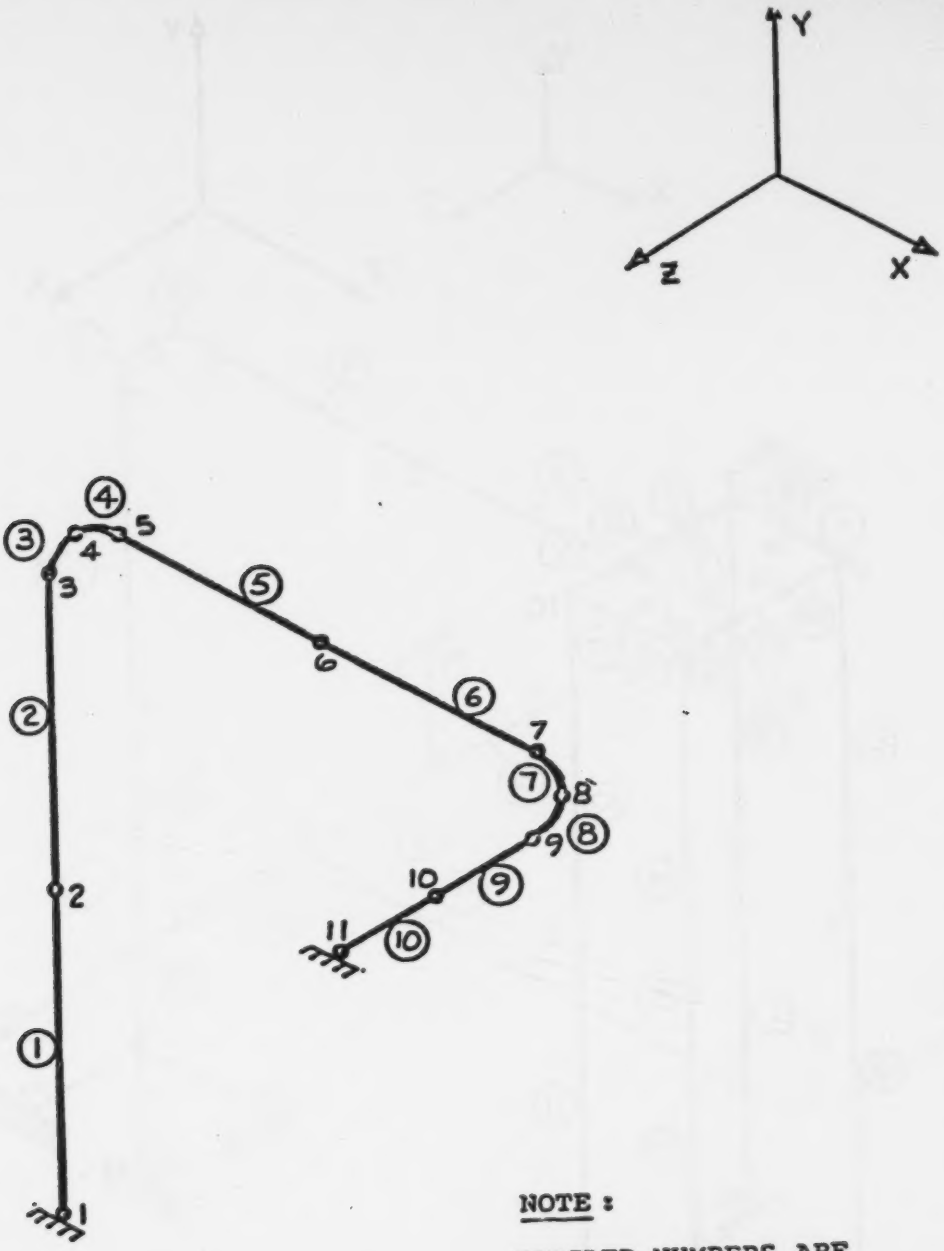
- A REPORT DESCRIBING THESE BENCHMARK PROBLEMS NUREG CR-1677 WAS ISSUED.

PROBLEM ENCOUNTERED WITH BENCHMARK PROBLEMS

1) SOME APPLICANT CODES DO NOT HAVE LARGE CAPACITY

2) ROTARY INERTIA AFFECTS ARE NEGLECTED

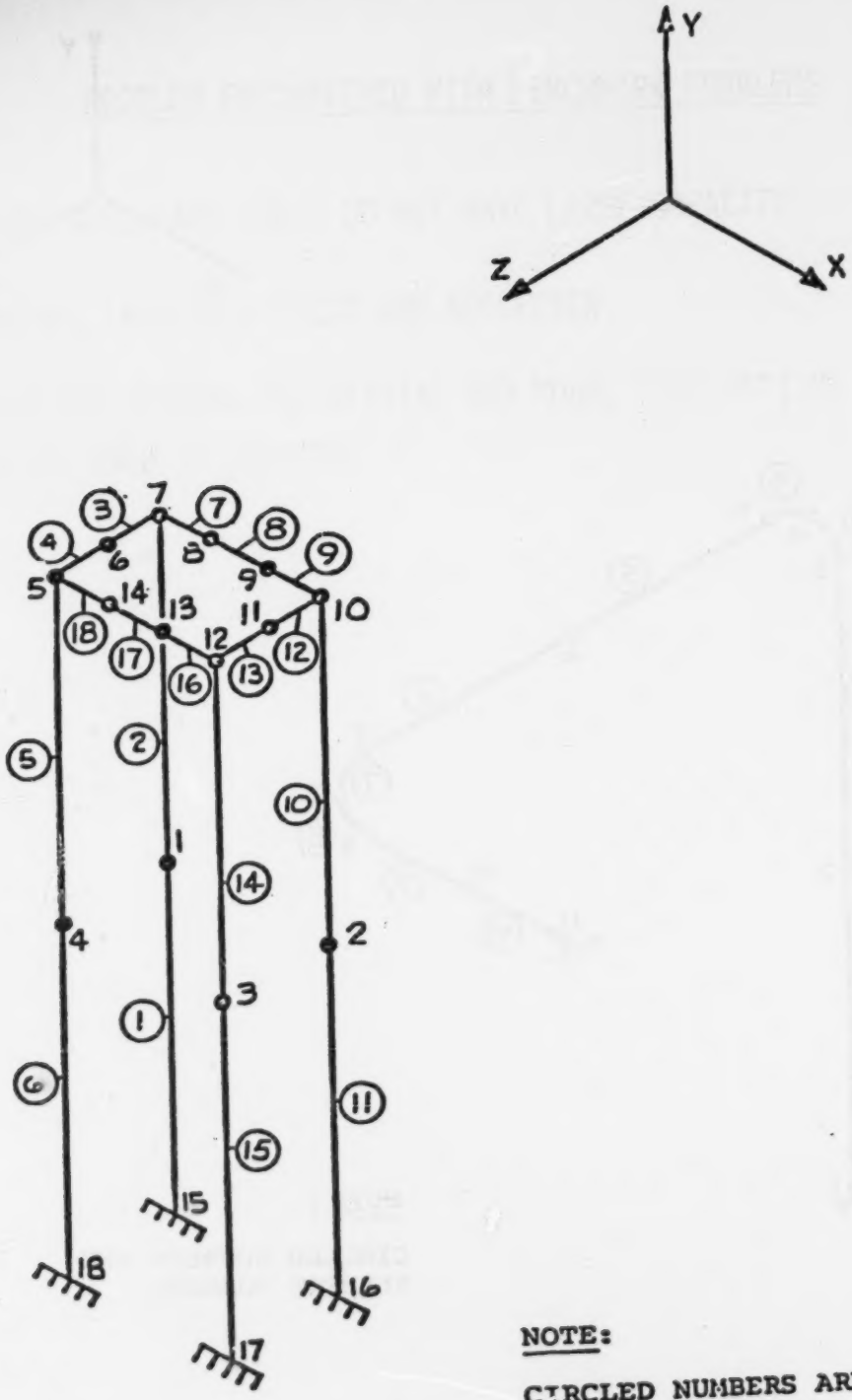
3) VARIOUS METHODS FOR SPATIAL AND MODAL COMBINATIONS BEING USED IN INDUSTRY



NOTE :

CIRCLED NUMBERS ARE ELEMENT NUMBERS.

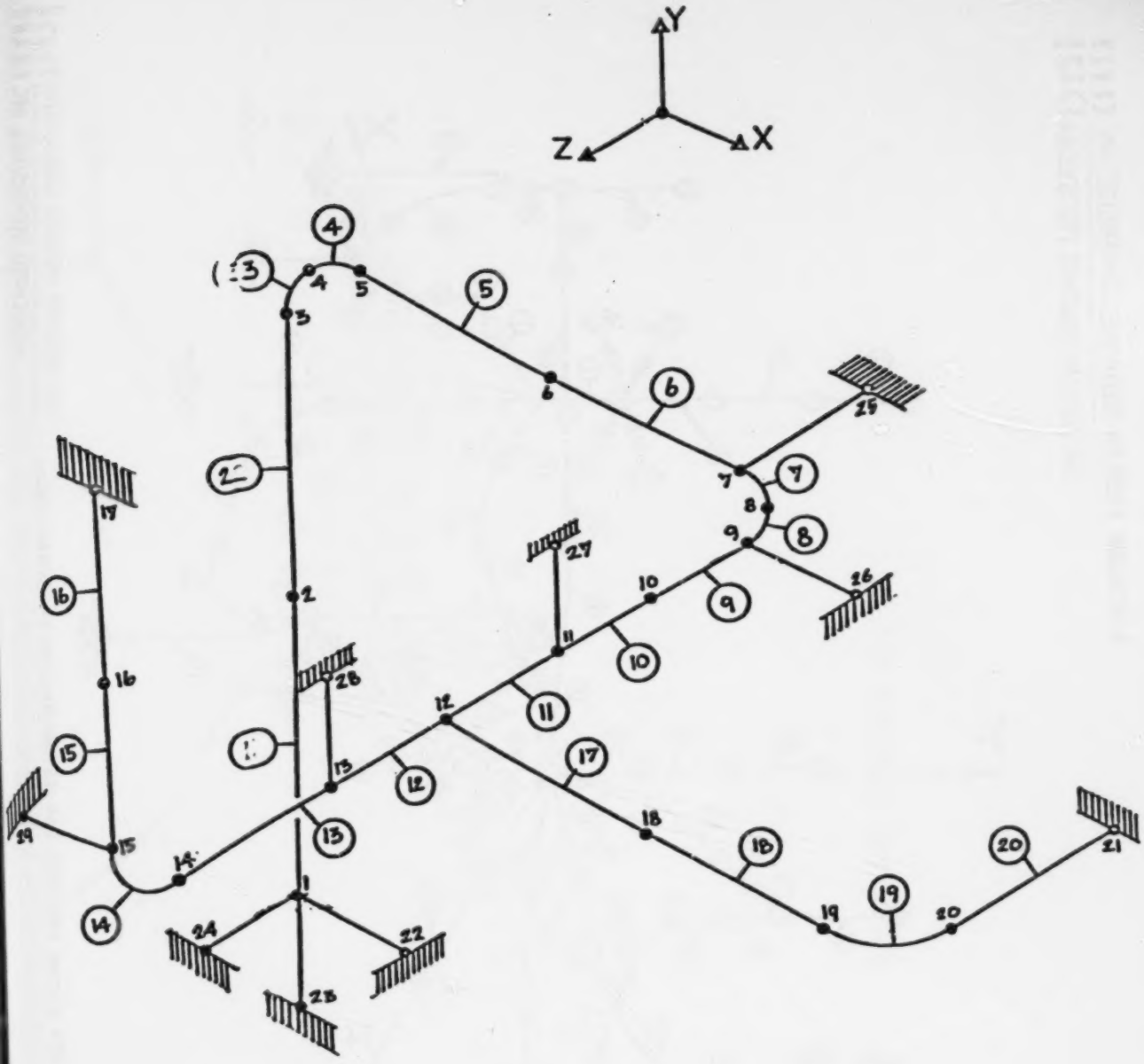
BENCHMARK PROBLEM NO. 1



NOTE:

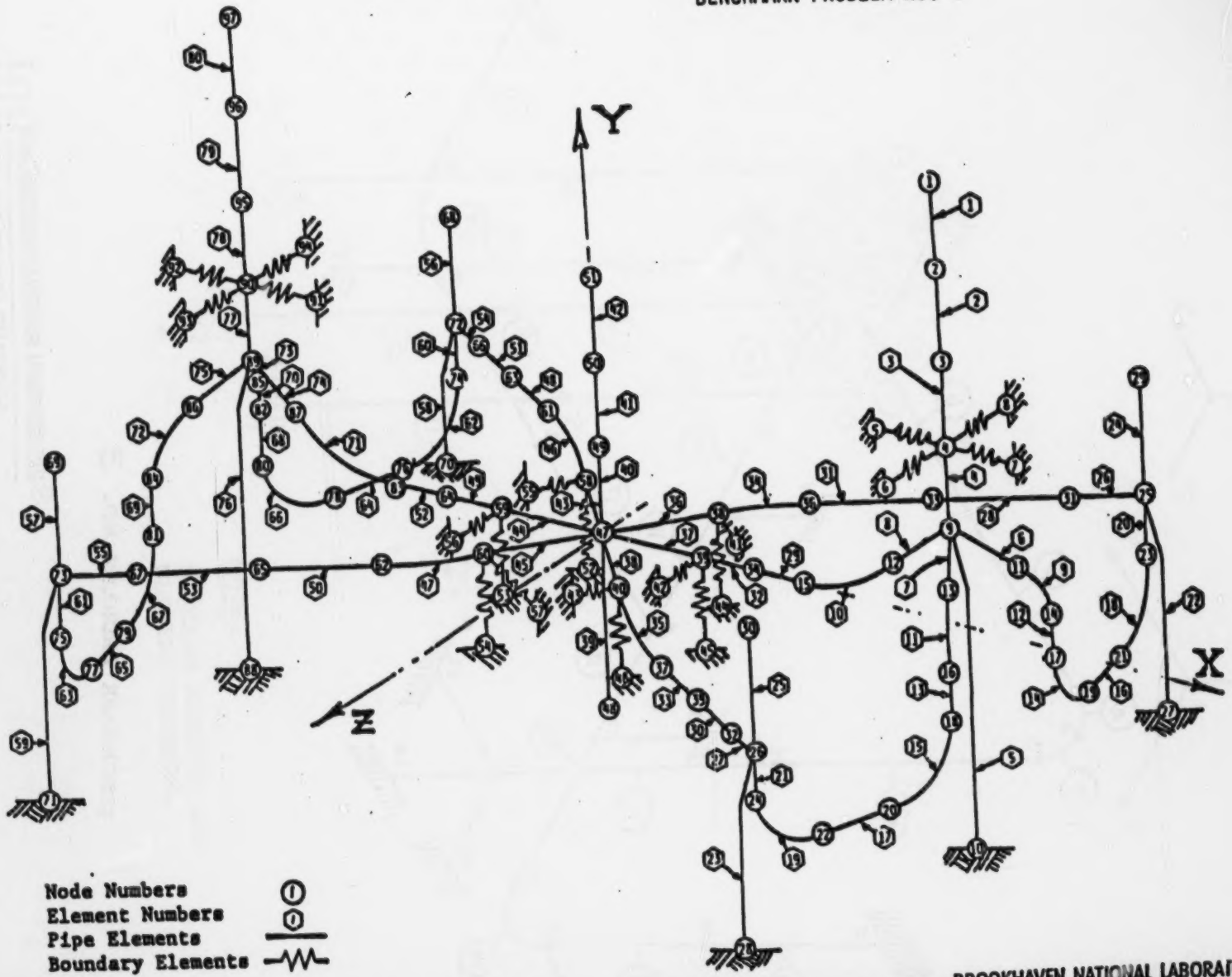
CIRCLED NUMBERS ARE ELEMENT NUMBERS.

BENCHMARK PROBLEM NO. 2



BENCHMARK PROBLEM NO. 3

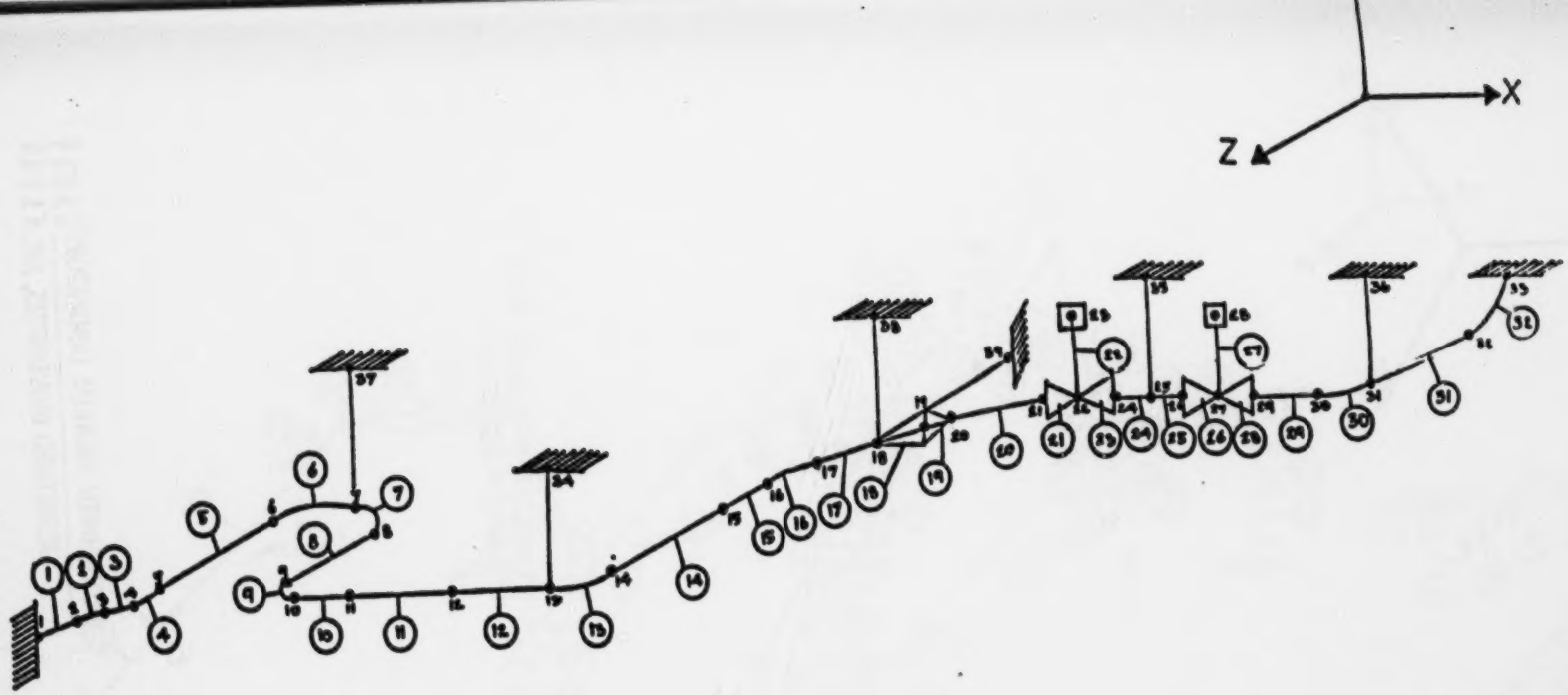
BENCHMARK PROBLEM NO. 4



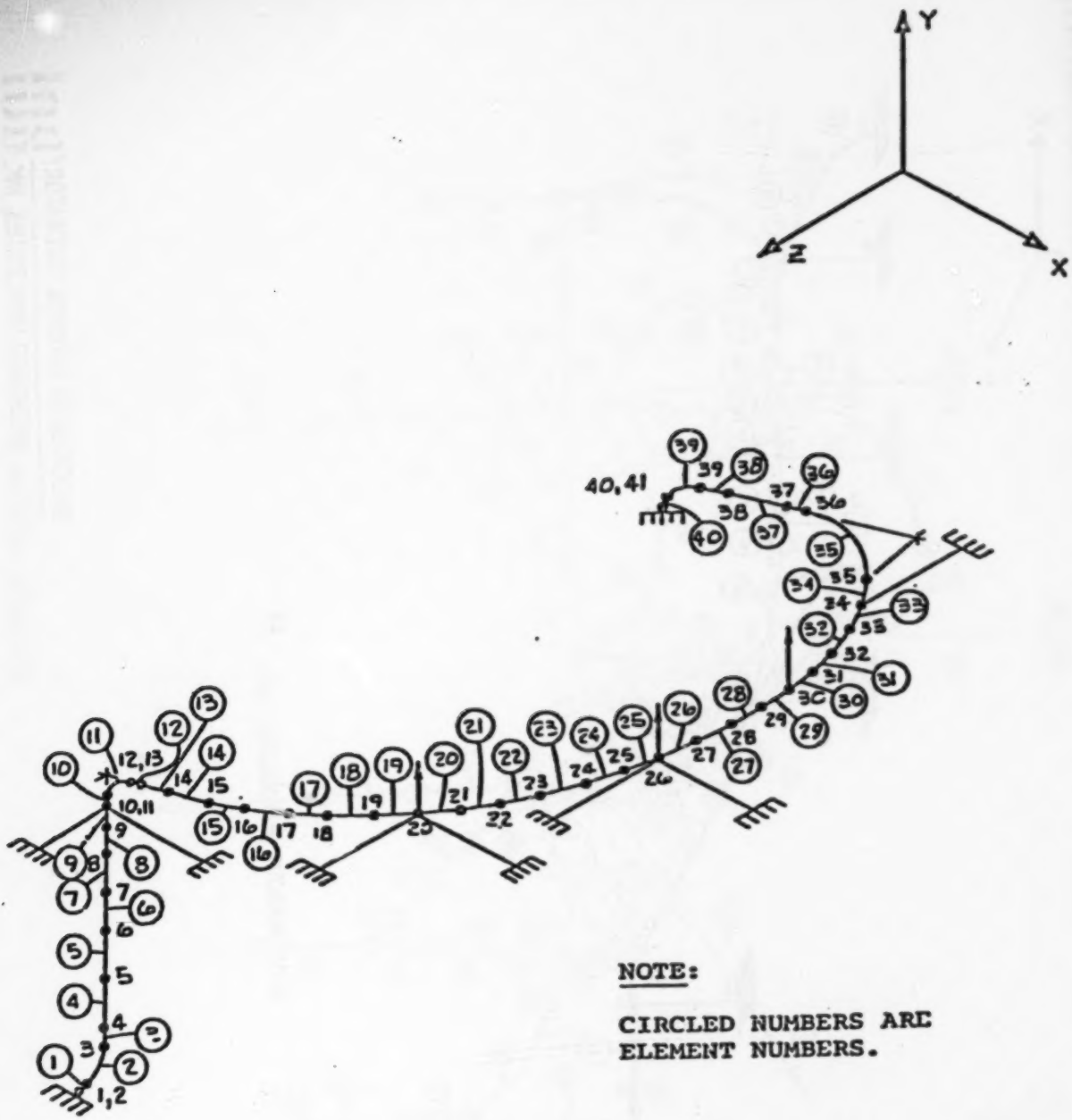
Node Numbers
Element Numbers
Pipe Elements
Boundary Elements

Note: Elements 22, 23, 58, and 59 are shown crooked for illustration purposes only.

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.



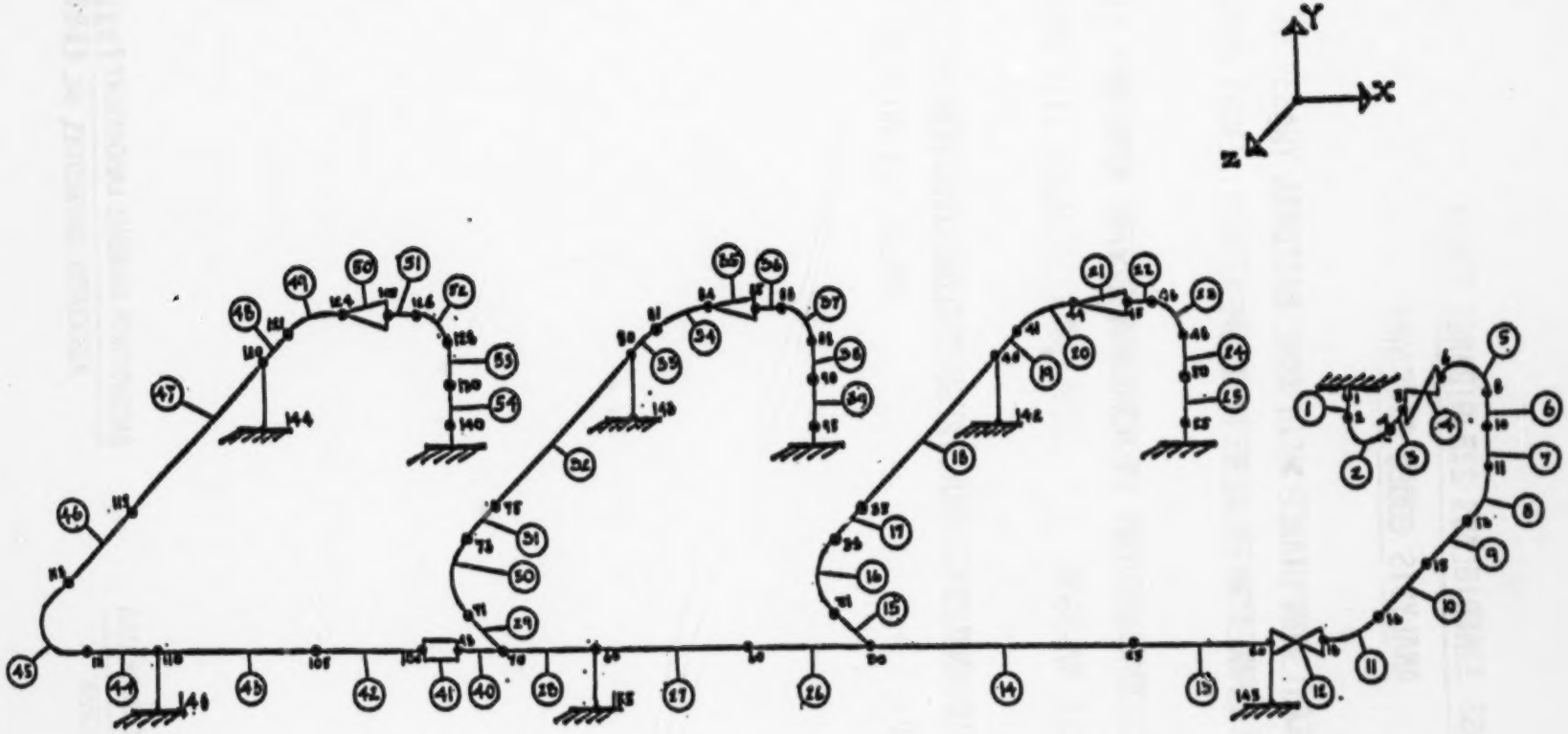
BENCHMARK PROBLEM NO. 5



NOTE:

CIRCLED NUMBERS ARE ELEMENT NUMBERS.

BENCHMARK PROBLEM NO. 6



BENCHMARK PROBLEM NO. 7

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

CLASS 1 AND CLASS 2,3 PIPING

ANALYSIS CODES

TO VERIFY APPLICANT COMPLIANCE WITH ASME CRITERIA TWO COMPUTER CODES, PSAFE1 AND PSAFE2 WERE DEVELOPED.

PSAFE1 PERFORMS STRESS EVALUATIONS IN ACCORDANCE WITH ASME-BP SECTION III, SUBARTICLE NB-3600.

PSAFE2 PERFORMS ELASTIC ANALYSIS AND STRESS EVALUATIONS PER ASME SECTION III NC/ND.

CLASS 1 AND CLASS 2,3 PIPING

ANALYSIS CODES

TO VERIFY APPLICANT COMPLIANCE WITH ASME CRITERIA TWO COMPUTER CODES, PSAFE1 AND PSAFE2 WERE DEVELOPED.

PSAFE1 PERFORMS STRESS EVALUATIONS IN ACCORDANCE WITH ASME-BP SECTION III, SUBARTICLE NB-3600.

PSAFE2 PERFORMS ELASTIC ANALYSIS AND STRESS EVALUATIONS PER ASME SECTION III NC/ND.

PSAFE1

A POST PROCESSOR DESIGNED TO PERFORM THE STRESS EVALUATION OF CLASS 1 PIPING IN ACCORDANCE WITH ASME SECTION III, SUBARTICLE NB-3600, 1974 ED.

SPECIFICALLY CODE PERFORMES CHECK FOR:

1) COMPONENT BOUNDARY PER NB-3641
2) MEMBRANE FAILURE PER EQUATION 9
3) ELASTIC CYCLING FATIQUE PER EQUATION 10
4) PLASTIC CYCLING FATIQUE PER EQUATION 11
   (IF NECESSARY)
5) PLASTIC HINGE FORMATION PER EQUATION 12
   (IF NECESSARY)
6) PLASTIC CYCLE LIMIT PER B 31.7 OR EQUATION 13
7) MAXIMUM ALTERNATING STRESS FOR ANY LOAD PAIR PER
   EQUATION 14
8) USAGE FACTOR USING APPROPRIATE S-N CURVES AND
   MINERS RULE

PSAFE2

- AN UPDATED VERSION OF EPIPE
- COULD BE USED FOR STRESS ANALYSIS SIMILAR TO EPIPE OR FOR A SPECIFIC ASME CLASS 2/3 EVALUATION OF PIPING SYSTEM
- ADDITIONAL CAPABILITIES:

  1) PERFORMS BOTH STATIC AND DYNAMIC ANALYSIS TOGETHER.
  2) PERFORMS S.A.M. ANALYSIS ALONG WITH THE STATIC AND DYNAMIC ANALYSES.
  3) MODIFIES STIFFNESS MATRIX FOR DYNAMIC CASES BY INTRODUCING NEW ELEMENTS TO SIMULATE SNUBBERS.
  4) STORES PIPE ELEMENT MOMENT DATA FOR VARIOUS LOADING CONDITIONS.
  5) PERFORMS CLASS 2/3 EVALUATION PER ASME SECTION III NC/ND, 1974 ED.
  6) SIMPLIFIES NODAL TEMERATURE AND ELEMENT PRESSURE DATA INPUT.

- LOADING CONDITIONS (IN GENERAL):

  1) INTERNAL AND EXTERNAL PRESSURE
  2) WEIGHT
  3) EARTHQUAKE
  4) THERMAL EFFECTS

ACCOMPLISHMENTS

1) BROUGHT BOTH PSAFE1 AND PSAFE2 TO FULL OPERATIONAL STATUS AND MADE THEM AVAILABLE TO NRC CONTRACTORS.

2) ISSUED DRAFT COPIES OF USER'S MANUALS FOR PSAFE1 AND PSAFE2

3) IMPLEMENTED LEVEL 2 UPDATE IN PSAFE2

4) INCORPORATED BANDWIDTH MINIMIZATION ALGORITHM IN PSAFE2

5) CONDUCTED WORKSHOP ON PSAFE1,2

6) ASSISTED NRC CONTRACTORS IN PROPER USE OF PSAFE1,2

ELASTIC PIPING ANALYSIS CODE

TO DEVELOPE THE BENCHMARK PROBLEMS IT WAS NECESSARY TO DEVELOPE AND PROOF TEST A ELASTIC PIPING ANALYSIS CODE. THE CODE DEVELOPED WAS EPIPE.

EPIPE CAPABILITIES

- STATIC ANALYSIS:

  SUSTAINED LOADS: GRAVITY, CONCENTRATED LOADS
  THERMAL LOADS
  ANCHOR/SUPPORT MOMENTS
  PRESSURE LOADS

- DYNAMIC ANALYSIS:

  EIGENVALUE SOLUTION
  TIME HISTORY BY MODAL SUPERPOSITION
  RESPONSE SPECTRUM ANALYSIS
  DIRECT STEP-BY-STEP INTEGRATION

- PIPE STRESS CALCULATIONS
- AUTOMATIC BANDWIDTH MINIMIZATION
- DATA CHECK OPTION
- NO LIMIT ON THE SIZE OF THE PROBLEM
- CONSISTS OF PIPE ELEMENTS (BOTH TANGENT AND BEND) AND BOUNDARY ELEMENTS (SPRING ELEMENT).

EPIPE (CON'T)

- MATERIAL PROPERTIES AS A FUNCTION OF TEMPERATURE.
- FOR RESPONSE SPECTRUM ANALYSIS: (FOR BOTH DISPLACEMENTS AND FORCES).
  - MODAL COMBINATIONS: SRSS
  - COMPONENT COMBINATION:

    A) ABSOLUTE SUM

    B) SRSS WITH CLUSTERING (REG. GUIDE 1.92)

    C) ABSOLUTE SUM WITH CLUSTERING

    D) ALGEBRAIC SUM WITH CLUSTERING

    E) MIXED RULE WITH CLUSTERING

    (I.E., ABSOLUTE OF Y IS SUMMED DIRECTLY WITH SRSS COMBINATION OF X AND Z).

ACCOMPLISHMENTS

1) BROUGHT CODE TO FULL OPERATIONAL STATUS AND MADE IT AVAILABLE TO NRC CONTRACTORS

2) ISSUED REVISED EPIPE USER'S MANUAL

3) UPDATED CODE TO ACCOMODATE CONFIRMATORY ANALYSIS TASK

4) INCORPORATED AN OPTION PERMITTING MULTIPLE SUPPORT EXCITATION

5) EXPANDED AND OPTIMIZED ELEMENT OUTPUT OPTIONS

6) ASSISTED NRC CONTRACTORS IN PROPER USE OF EPIPE

EVALUATION OF APPLICANT ANALYSIS METHODS

NEW AND ALTERNATE ANALYSIS PROCEDURES MUST BE VERIFIED AND WHERE APPROPRIATE BENCHMARK PROBLEMS AND SOLUTIONS DEVELOPED.

ACCOMPLISHMENTS

1) EVALUATED THE PSEUDO FORCE METHOD OF NONLINEAR ANALYSIS. ISSUED A REPORT SUMMARIZING FINDINGS.

2) EVALUATED THE SUBSYSTEM METHOD OF ELASTIC PIPING SYSTEM ANALYSIS. ISSUED A REPORT SUMMARIZING FINDINGS.

CONCLUSIONS

1) THE PSEUDO FORCE METHOD IS A CORRECT AND VIABLE ALTERNATE PROCEDURE TO PERFORM NONLINEAR ANALYSIS WHERE THE NUMBER OF NONLINEAR ELEMENTS ARE DISCRETELY LOCATED AND LIMITED IN NUMBER.

2) THE SUBSYSTEM ANALYSIS METHOD HAS NO THEORETICAL BASIS. ACCEPTABLE RESULTS HOWEVER CAN BE ACHIEVED WITH THE METHOD IF THE NRC RESTRICTIONS CONCERNING ITS USE ARE FOLLOWED.

SNUBBER RESEARCH AND TESTING

Armand T. Onesto

(Energy Technology Engineering Center)

1.0 Introduction

This paper summarizes the results of three separate snubber programs performed for the Nuclear Regulatory Commission (NRC) by the Energy Technology Engineering Center (ETEC). The snubber programs which were performed for different NRC divisions collectively represent a broad based program providing the NRC with valuable experience pertaining to analysis, design, application, inspection and testing of safety related snubbers.

The three programs discussed herein are:

1) Snubber Sensitivity Study (MEB)
2) Single Versus Multiple Snubber Test (OSD)
3) Snubber Application and Test (MERB)

The Snubber Sensitivity Study and the Single Versus Multiple Snubber Test programs were completed during FY80. The Snubber Application and Test program was initiated in FY79 and will continue through FY81. The objectives and the results of the two completed programs and the partially completed program will be discussed along with a brief description of the remaining tasks.

2.0 Discussion

2.1 Snubber Sensitivity Study - The object of this program was to identify snubber characterization parameters that affect system response, evaluate response sensitivity to the identified parameters and formulate the findings into simplified design and analysis guidelines that would bound system response within acceptable limits.

Analytical studies of simple structural systems were performed to determine snubber response sensitivity to clearance, stiffness, activation level, release rate and friction. The acceleration activated mechanical snubber and the velocity activated (poppet-type) hydraulic snubber were studied. These studies provided guidance for establishing acceptable limits for the parameters affecting system response. Guidelines for simplified design and analysis of snubber supported systems including snubber selection, support stiffness requirements and operational limits for snubber parameters were provided.

2.2 Single Versus Multiple Snubber Test - The object of this test was to determine the effect of replacing a single large snubber with two smaller snubbers of approximately the same total capacity and to determine the effects of mismatch of snubbers used as pairs. Pacific Scientific mechanical snubbers and Grinnell hydraulic snubbers were used for testing.

A simulated piping system was used in place of a prototypical line in order to provide more accurate control of parameters affecting system response. The test system consisted of a 10 foot long carbon steel tube (6 1/2 inch outside diameter, 0.5 inch wall) which was cantilevered from a

hydraulically activated shaker table. The beam was oriented parallel to the ground and supported a 1000 pound lead maas at the free end and a trapeze structure 4 1/2 feet from the base. Snubbers were attached between the trapeze structure and a strongback structure which was attached to a 50,000 pound seismic mass. The test system was designed and instrumented so that the effects of input (sine or seismic), support type (single or pair), supports (rigid strut, mechanical or hydraulic snubber), mismatch of end fitting clearance, activation level or release rate on load sharing of snubber pairs, system response and/or piping stresses could be studied.

The results of the test indicated that load sharing is affected more by mismatch of end fitting clearance than activation level and/or release rate. Uniform load sharing (±5%) can be expected if end fitting clearance mismatch is less than 0.01 inches and activation level and release rate are between 8 to 25 inches/minute and 4 to 15 inches/minute respectively. However, load sharing of 30-70% could be expected if the end fitting clearance mismatch is 0.05 inches.

During testing, ovalization of the snubber clevis pin inserts created problems. The excess clearance resulting from the ovalization produced uneven load sharing and unanticipated load variations. Replacement of worn hardware was required periodically during testing. In light of these observations, periodic inservice inspection of end fitting hardware may be warranted.

2.3 Snubber Application and Test Program - The object of this program is to develop a background support document for a proposed Regulatory Guide on snubber Application and Test Requirements (Criteria Support Document). The following procedure will be used to meet program objectives:

1) Identify existing Snubber criteria.
2) Supplement, modify or delete criteria.
3) Compile existing criteria related data.
4) Perform additional analysis or tests.
5) Formulate Criteria Support Document.

Items 1 through 3 have been completed and the detailed test and analytical studies have been proposed. Assistance from industry to establish existing and necessary criteria was obtained from a Survey Questionnaire which was sent to snubber manufacturers, nuclear plants, NSSS, and A&E's. The results were used to assist in the formulation of the remainder of the program.

3.0 Summary

The Snubber Sensitivity Study and the Single Versus Multiple Snubber Test programs have yielded valuable insights regarding the operability of snubbers and their sensitivity to variations in characterization parameters. These insights will provide valuable inputs to the Snubber Application and Test Program. At the completion of the snubber programs it is intended that many of the significant questions regarding the design, application, analysis, inspection and testing of snubbers shall be answered and documented in a single source.

SNUBBER RESEARCH AND TESTING

BY

ARMAND T. ONESTO

(ENERGY TECHNOLOGY ENGINEERING CENTER)

* * * SNUBBER RESEARCH AND TESTING * * *

- PRESENTATION OBJECTIVE
  - DEFINE SNUBBER PROGRAM OBJECTIVES
    - SNUBBER SENSITIVITY STUDY (MEB)
    - SINGLE VERSUS MULTIPLE SNUBBER TEST (OSD)
    - SNUBBER APPLICATION AND TEST (MERB)
  - DISCUSS PROGRAM METHODOLOGY / RESULTS
    - SNUBBER SENSITIVITY STUDY
    - SINGLE VERSUS MULTIPLE SNUBBER TEST
  - FORMULATION OF SNUBBER APPLICATION AND TEST PROGRAM
    - IMPACT OF COMPLETED STUDIES
    - SNUBBER SURVEY QUESTIONNAIRE
    - PROPOSED STUDIES
  - SUMMARIZE STATUS OF SNUBBER RESEARCH

* * * SNUBBER RESEARCH AND TESTING * * *

- PROGRAM OBJECTIVES
  - SNUBBER SENSITIVITY STUDY
    - IDENTIFY SNUBBER PARAMETERS THAT EFFECT RESPONSE
    - EVALUATE RESPONSE SENSITIVITY TO IDENTIFIED PARAMETERS
    - DEVELOP SIMPLIFIED DESIGN AND ANALYSIS GUIDELINES
  - SINGLE VERSUS MULTIPLE SNUBBER TEST
    - DETERMINE EFFECT OF REPLACING A SINGLE LARGE SNUBBER WITH TWO SMALLER SNUBBERS
    - EVALUATE EFFECTS OF MISMATCH FOR SNUBBER PAIR
      - END FITTING CLEARANCE
      - ACTIVATION LEVEL
      - RELEASE RATE
  - SNUBBER APPLICATION AND TEST
    - DEVELOP BACKGROUND SUPPORT DOCUMENT FOR PROPOSED REGULATORY GUIDE ON SNUBBER APPLICATION AND TEST REQUIREMENTS

*** SNUBBER RESEARCH AND TESTING ***

- SNUBBER SENSITIVITY STUDY
  - ANALYTICAL EFFORT
    - STIFFNESS
    - CLEARANCE
    - ACTIVATION LEVEL
    - RELEASE RATE
    - FRICTION
  - SIMPLIFIED DESIGN AND ANALYSIS PROCEDURES
    - SNUBBER SELECTION
    - STIFFNESS REQUIREMENTS
    - ACTIVATION LEVEL RANGE
    - RELEASE RATE RANGE
    - CLEARANCE RANGE
    - FRICTION
    - LINEAR REPRESENTATION OF SNUBBER

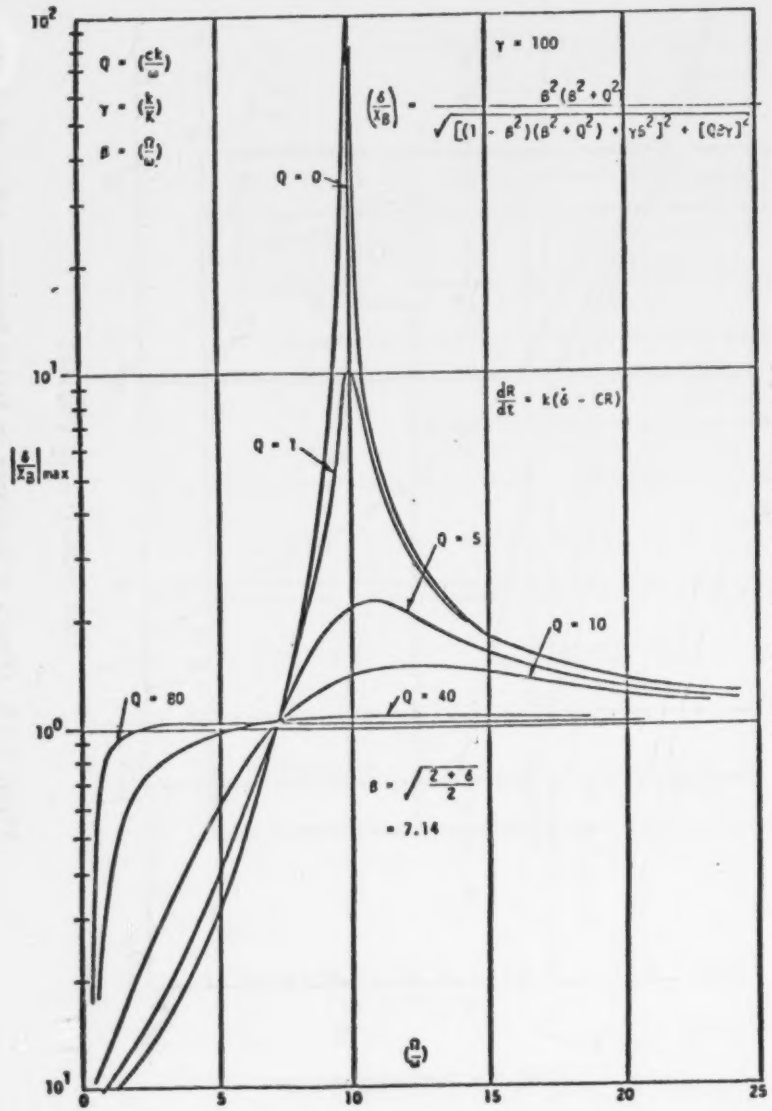


FIGURE A.6.1 SYSTEM RESPONSE FUNCTION (β)

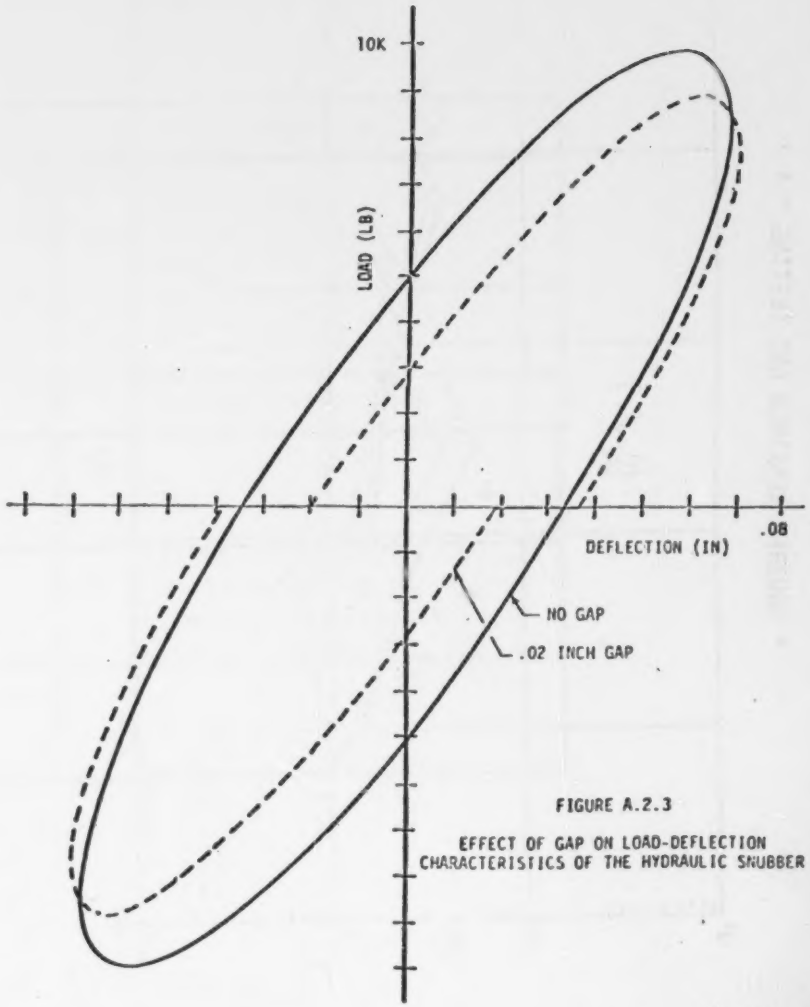


FIGURE A.2.3

EFFECT OF GAP ON LOAD-DEFLECTION CHARACTERISTICS OF THE HYDRAULIC SNUBBER

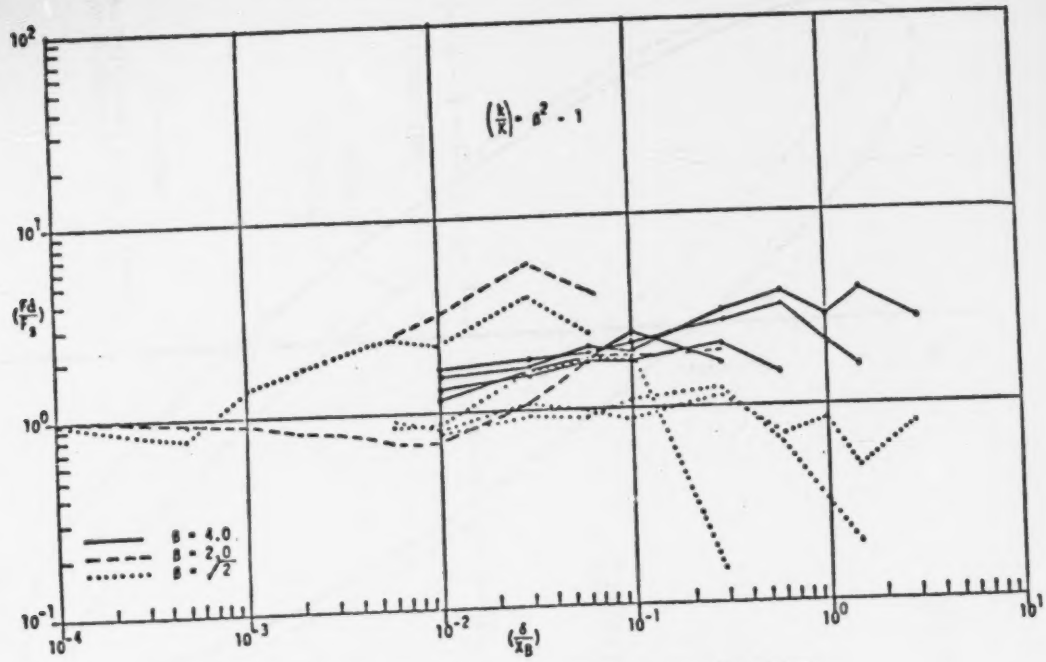


FIGURE A.8.3 MAXIMUM IMPACT LOAD FACTORS (TRANSIENT RESPONSE)

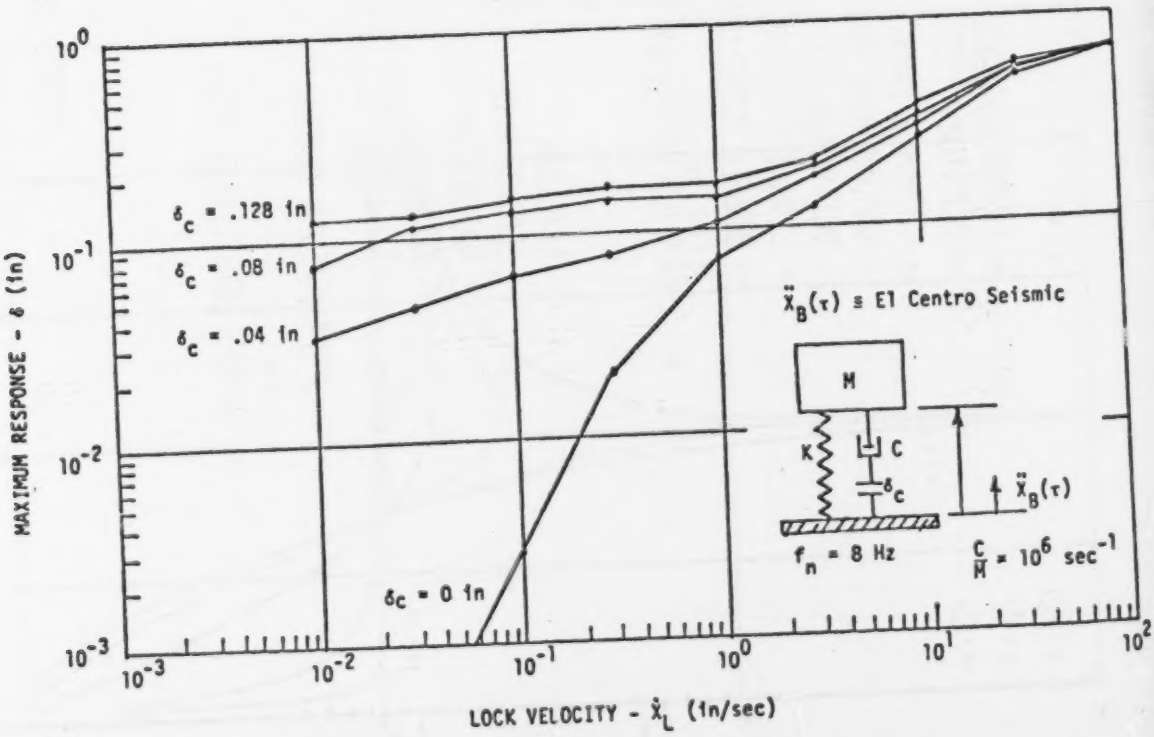


Figure 7.3.5 Effects of Clearance on System Response, $C/M = 10^6$ sec^{-1}

ALLOWABLE AND DESIRABLE SNUBBER RESPONSE PARAMETER RANGES

PARAMETER	TYPE	ALLOWABLE RANGE	DESIRABLE RANGE
ACTIVATION LEVEL	H	< 50 in/min if $\dot{X}_B$ > 5 in/min < 70 in/min if $\dot{X}_B$ < 5 in/min	Low as possible, but great enough to preclude lock-up during thermal expansion.
	M	< .08 g	
RELEASE RATE ($\dot{X}_B$)	H	< .07 x ($\frac{\text{Rated Load}}{\text{Component Wt}}$) in/min	5 → 30 in/min
	M	See Activation Level	See Activation Level
CLEARANCE	B	< .05 in.	0
FRICTION	B	< 40% Rated Load	As large as possible but small enough to maintain thermal stresses within acceptable limits.
STIFFNESS	B	> $\frac{20}{\alpha - 20\alpha_H}$	>> $\frac{20}{\alpha - 20\alpha_H}$

NOTES:

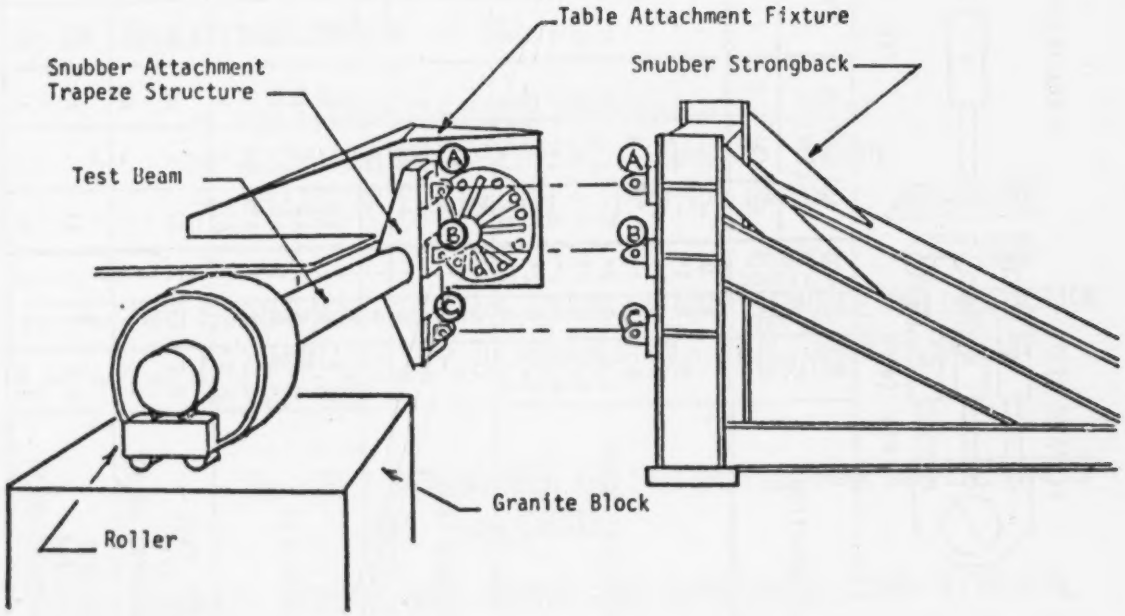
H = HYDRAULIC
M = MECHANICAL
B = BOTH

α = Flexibility of pipe @ snubber location (in/lb)

α_H = Combined flexibility of all hardware between the center of the component and "ground" (in/lb)

* * * SNUBBER RESEARCH AND TESTING * * *

- SINGLE VERSUS MULTIPLE SNUBBER TEST
  - TEST SNUBBERS
    - HYDRAULIC (GRINNELL 1 1/2 AND 2 1/2)
    - MECHANICAL (PACIFIC SCIENTIFIC PSA-3 AND PSA-10)
  - TEST SYSTEM (SIMULATED PIPING LOOP)
    - 4 HZ NATURAL FREQUENCY (W/O SNUBBERS)
    - 8 HZ NATURAL FREQUENCY (WITH SNUBBERS)
    - 15K-20K SNUBBER LOADS
    - HIGH PIPING STRESSES (50,000 PSI)
    - LARGE DEFLECTIONS (.1 - .5 IN. FREE RESPONSE)
    - RESPONSE PARAMETER CONTROL
    - DISPLACEMENT AND INERTIAL CONTROLLED LOADINGS
  - TEST VARIABLES
    - INPUT LOADS
      - SEISMIC
      - SINE
    - SUPPORT CONFIGURATION
      - SINGLE
      - PAIR
    - SUPPORT TYPE
      - RIGID STRUT
      - HYDRAULIC SNUBBER
    - END FITTING CLEARANCE
    - ACTIVATION LEVEL
    - RELEASE RATE



Attachment Location X: Ⓧ

Snubber Attachment Locations

Single Vs Multiple Snubber Tests

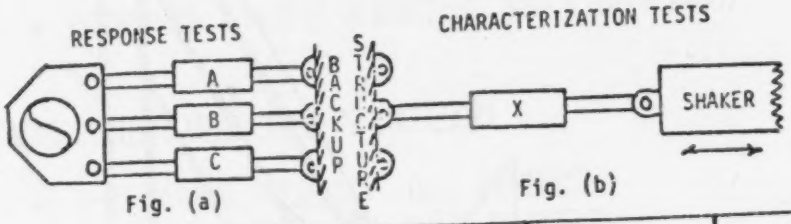


Fig. (a) Fig. (b)

TEST NUMBER	TEST DESCRIPTION	LOCATION	TYPE OF TEST	NUMBER OF SNUBBERS	ACTIVATION LEVEL			RELEASE RATE			PIN CLEARANCE		
					A	B	C	A	B	C	A	B	C
1	SYSTEM CHARACTERIZATION	-	R	0	-	-	-	-	-	-	-	-	-
2		B	R	1*	-	0.	-	-	0.	-	-	.00	-
3		B	R	1*	-	0.	-	-	0.	-	-	.02	-
4		B	R	1*	-	0.	-	-	0.	-	-	.05	-
5		A,C	R	2*	0.	-	0.	0.	-	0.	.00	-	.00
6		A,C	R	2*	0.	-	0.	0.	-	0.	.00	-	.02
7		A,C	R	2*	0.	-	0.	0.	-	0.	.00	-	.05
8		A,C	R	2*	0.	-	0.	0.	-	0.	.02	-	.02
9		A,C	R	2*	0.	-	0.	0.	-	0.	.05	-	.05
10	SINGLE SNUBBER TEST	X	C	1	-	-	-	-	-	-	-	-	-
11		B	R	1	-	A1	-	-	R1	-	-	.00	-
12		B	R	1	-	A1	-	-	R1	-	-	.02	-
13		B	R	1	-	A1	-	-	R1	-	-	.05	-
14	REVERSED SNUBBER	B	R	1	-	TBD	-	-	TBD	-	-	.00	-
15,16	MATCHED PAIR TEST	X	C	2	-	-	-	-	-	-	-	-	-
17		A,C	R	2	A1	-	A1	R1	-	R1	.00	-	.00
18		A,C	R	2	A1	-	A1	R1	-	R1	.00	-	.02
19		A,C	R	2	A1	-	A1	R1	-	R1	.00	-	.05
20		A,C	R	2	A1	-	A1	R1	-	R1	.02	-	.02
21		A,C	R	2	A1	-	A1	R1	-	R1	.05	-	.05
22	ACTIVATION LEVEL MISMATCH TEST (HYDRAULIC SNUBBERS ONLY)	X	C	1	-	-	-	-	-	-	-	-	-
23		A,C	R	2	A1	-	A2	R1	-	R1	.00	-	.00
24		X	R	1	-	-	-	-	-	-	.00	-	.00
25		A,C	R	2	A1	-	A3	R1	-	R1	.00	-	.00
26	RELEASE RATE MISMATCH TEST (HYDRAULIC SNUBBERS ONLY)	X	C	1	-	-	-	-	-	-	-	-	-
27		A,C	R	2	A1	-	A1	R1	-	R2	.00	-	.00
28		X	C	1	-	-	-	-	-	-	.00	-	.00
29		A,C	R	2	A1	-	A1	R1	-	R3	.00	-	.00

* = Rigid Strut(s)
C = Characterization
R = Response Tests
X = Characterization Fixture
TBD = To be determined
A1, A2, A3 = Activation Levels
R1, R2, R3 = Release Rates

RESPONSE TESTS:
a) Low Level Sine
b) High Level Sine
c) Seismic Simulation

CHARACTERIZATION:
a) Activation Level
b) Release Rate
c) Lost Motion
d) Stiffness

* * * SNUBBER RESEARCH AND TESTING * * *

- SUMMARY OF RESULTS OF SINGLE VS MULTIPLE SNUBBER TEST
  - UNIFORM LOAD SHARING ($\pm$ 5%)
    - END FITTING CLEARANCE DIFFERENTIAL .01 IN
    - ACTIVATION LEVEL BETWEEN (8 ± 2) AND 25 IN/MIN
    - RELEASE RATE BETWEEN (4 ± 1) AND 15 IN/MIN
  - LOAD SHARING MORE SENSITIVE TO END FITTING CLEARANCE AND LOST MOTION DIFFERENTIALS THAN TO DIFFERENTIALS IN ACTIVATION LEVEL OR RELEASE RATE
    - 30 - 70 % SHARING WHEN END FITTING CLEARANCE DIFFERENTIAL OF .05 INCH EXISTS
  - HYDRAULIC SNUBBERS HAVE GREATER LOAD DIFFERENTIALS THAN MECHANICAL SNUBBERS FOR A GIVEN CLEARANCE DIFFERENTIAL FOR HARMONIC INPUT
  - MECHANICAL SNUBBERS HAVE GREATER LOAD DIFFERENTIALS THAN HYDRAULIC SNUBBERS FOR A GIVEN CLEARANCE DIFFERENTIAL FOR SEISMIC INPUT

* * * SNUBBER RESEARCH AND TESTING * * *

- SNUBBER APPLICATION AND TEST PROGRAM
  - COVERS ALL ASPECTS OF SNUBBER USAGE
    - SEISMIC RESTRAINTS
    - VIBRATION SUPRESSION
    - PIPE WHIP RESTRAINTS
    - OTHER
  - FOCUS ON SNUBBER CRITERIA
    - DESIGN CRITERIA
    - APPLICATION CRITERIA
    - ANALYSIS CRITERIA
    - INSERVICE TEST/INSPECTION CRITERIA
    - QUALIFICATION/ACCEPTANCE TESTING CRITERIA
  - COMPILATION OF EXISTING STUDIES
    - ANALYTICAL
    - TEST
    - HISTORICAL RECORDS
  - NEW TEST PROGRAMS
  - NEW ANALYTICAL STUDIES

* * * SNUBBER RESEARCH AND TESTING * * *

- PROGRAM OBJECTIVE
  - DEVELOP BACKGROUND DOCUMENT FOR "PROPOSED" REGULATORY GUIDE
    - ESTABLISH CRITERIA TO BE CONTAINED IN PROPOSED REGULATORY GUIDE
    - DETERMINE SPECIFIC PARAMETERS TO BE ADDRESSED
    - COMPILE EXISTING DATA FOR JUSTIFICATION OF ITEMS IDENTIFIED ABOVE
    - DEFINE REQUIREMENTS FOR ADDITIONAL ANALYSIS AND TESTS
    - CONDUCT REQUIRED STUDIES
      - ANALYTICAL
      - TEST
    - FORMULATE SUPPORT DOCUMENT

* * * SNUBBER RESEARCH AND TESTING * * *

- CRITERIA
  - SNUBBER APPLICATION CRITERIA
  - SNUBBER DESIGN CRITERIA
  - SNUBBER ANALYSIS CRITERIA
  - SNUBBER IN-SERVICE INSPECTION CRITERIA
  - SNUBBER QUALIFICATION CRITERIA

* * * SNUBBER RESEARCH AND TESTING * * *

- INDUSTRY ASSISTANCE
  - IDENTIFY CRITERIA
    - NEW CRITERIA (SUPPLEMENTAL)
    - CRITERIA TO BE DELETED
    - CRITERIA TO BE MODIFIED
    - EXISTING CRITERIA DOCUMENTS
  - IDENTIFY GENERIC SNUBBER PROBLEMS
  - FORMULATE TEST PROGRAM (MALFUNCTION STUDY)
    - IDENTIFY POTENTIAL SNUBBER FAILURE MODES
    - IDENTIFY SNUBBER RELATED FAILURES

* * * SNUBBER RESEARCH AND TESTING * * *

- MAJOR STUDIES
  - SNUBBER MALFUNCTION STUDY
  - EFFECTS OF DRAG LOAD ON PIPING STRESSES
  - SNUBBER SENSITIVITY
  - EVALUATION OF QUALIFICATION TEST REQUIREMENTS
  - MULTIPLE LOCATIONS FOR AXIAL RESTRAINTS
  - EVALUATION OF RESPONSE PARAMETERS FOR PIPE WHIP RESTRAINTS
  - EVALUATION OF EXISTING TEST DATA

* * * SNUBBER RESEARCH AND TESTING * * *

- MINOR STUDIES
  - BACKUP STIFFNESS REQUIREMENTS
  - SUPPORT-STRUCTURE INTERACTIONS
  - MULTIPLE SNUBBER USAGE
  - SIZE LIMITATIONS
  - SNUBBER ORIENTATION
  - SNUBBER PLACEMENT (REDUNDANCY)
  - RESPONSE PARAMETER LIMITS
  - FAILURE MODE REQUIREMENTS
  - END FITTING FLEXIBILITY AND CLEARANCES
  - SIMPLIFIED ANALYSIS REQUIREMENTS
  - LINEAR SNUBBER MODEL
    - STIFFNESS
    - CLEARANCE
    - DAMPING
  - BOUNDARY DEFINITION
  - ACCEPTANCE RANGE FOR SNUBBER DEGRADATION
  - 3-33 HZ
  - 350°F FOR 72 HOURS - QUALIFICATION REQUIREMENTS
  - HYDRAULIC FLUID PARTICLE SIZE REQUIREMENTS
  - ULTIMATE LOAD EVALUATED ANALYTICALLY
  - SCHEDULES/SAMPLES
    - PRE-SERVICE INSPECTION - 6 MONTHS
    - IN-SERVICE EXAMINATION FREQUENCY (SEE SCHEDULE)
    - SAMPLE SIZE REQUIREMENTS

* * * SNUBBER RESEARCH AND TESTING * * *

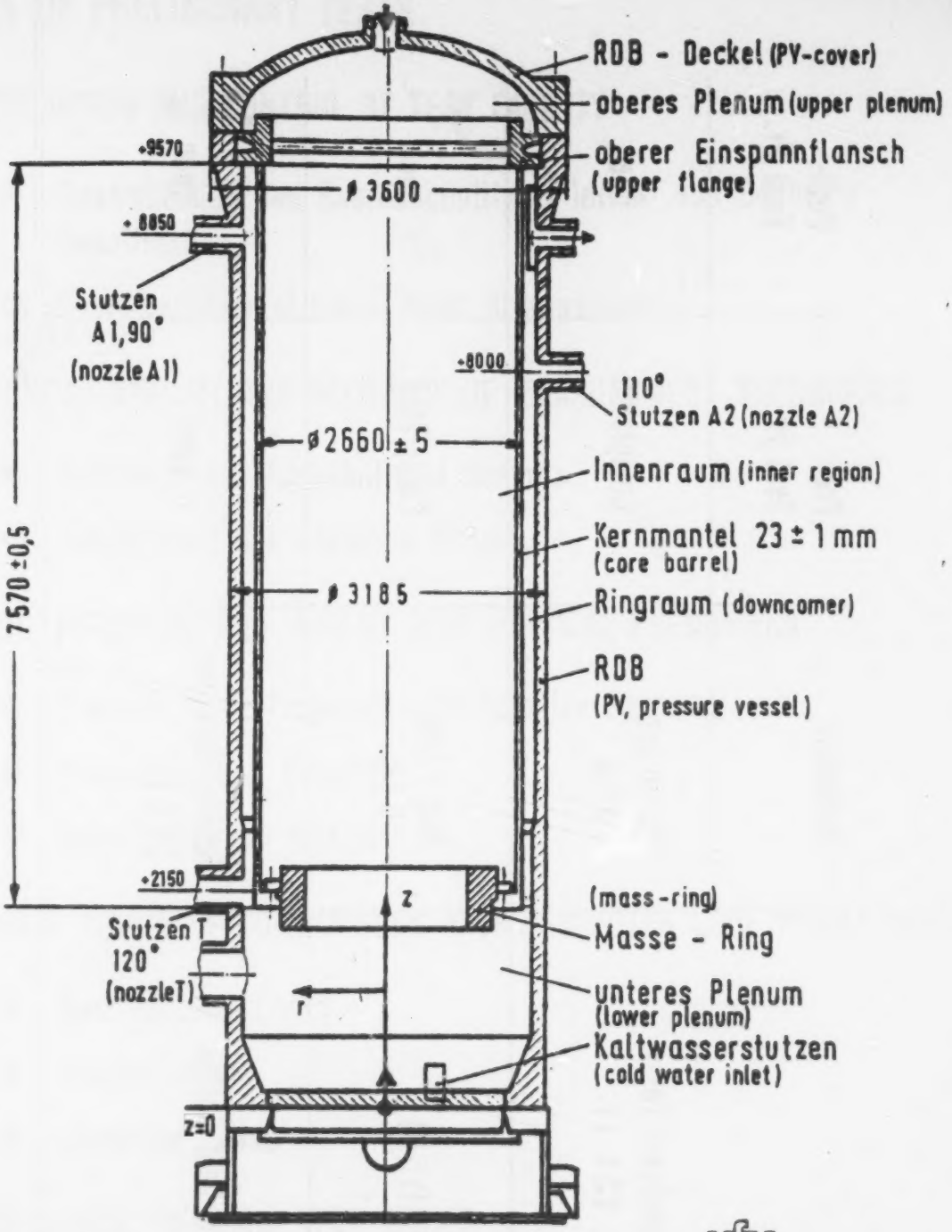
- SUMMARY OF COMPLETED WORK
  - SNUBBER RESPONSE SENSITIVITY
  - SIGNIFICANT SNUBBER RESPONSE PARAMETERS
  - PARAMETERS EFFECTING LOAD SHARING
  - EXISTING CRITERIA DOCUMENTS
  - INDUSTRY REQUIREMENTS/NEEDS
  - INSERT OVALIZATION
- REMAINING STUDIES
  - 7 MAJOR TASKS
  - 18 "MINOR"
- COMPLETION OF CRITERIA SUPPORT DOCUMENT
  - DESIGN
  - ANALYSIS
  - APPLICATION
  - INSPECTION
  - TEST

EXPERIMENTAL AND PRELIMINARY ANALYTICAL RESULTS OF COUPLED FLUID-STRUCTURE INTERACTIONS DURING BLOWDOWN OF THE HDR VESSEL

U. SCHUMANN (IRE)
L. WOLF (PHDR)

KERNFORSCHUNGSZENTRUM KARLSRUHE
GERMANY

EIGHTH WATER REACTOR SAFETY RESEARCH INFORMATION MEETING
GAITHERSBURG, OCTOBER 27-31, 1980



RDB und Kernmantelgeometrie für Blowdown Exp.

Anlage 1

RPV and Core-Barrel Geometries for Blowdown Experiments

	Schedule	Pre-Test Pr.	Post-Test Pr.
1. Preliminary Blowdown Tests (V29.2, V31, V31.1)	March-june 1980	Completed	open
2. Snapback Tests (V59)	Aug. / Sept. 1980	Completed	open
3. Main Blowdown Test Series (V32-V41)	Spring / Summer 1981	open	open

RPV-I Fluid-Structural Interactions Program at HDR

GOALS OF PRELIMINARY TESTS

— OPERATION AND CONTROL OF TEST FACILITY

- Controllability and Reproducibility of Initial- and Boundary Conditions
- Characteristics of Break Disks Arrangement

— FUNCTIONABILITY AND ACCURACY OF MEASUREMENT TECHNIQUES

- Reliability and Availability of Sensors
- Temperature and Pressure Effects

— IDENTIFICATION AND INSIGHT INTO PHYSICAL PHENOMENA

- Pressure Wave Propagation in Complex Geometry
- Fluid-Structural Coupling
- Non-Equilibrium Effects

— FIRST ROUND OF COMPARISONS WITH COMPUTER CODE PREDICTIONS

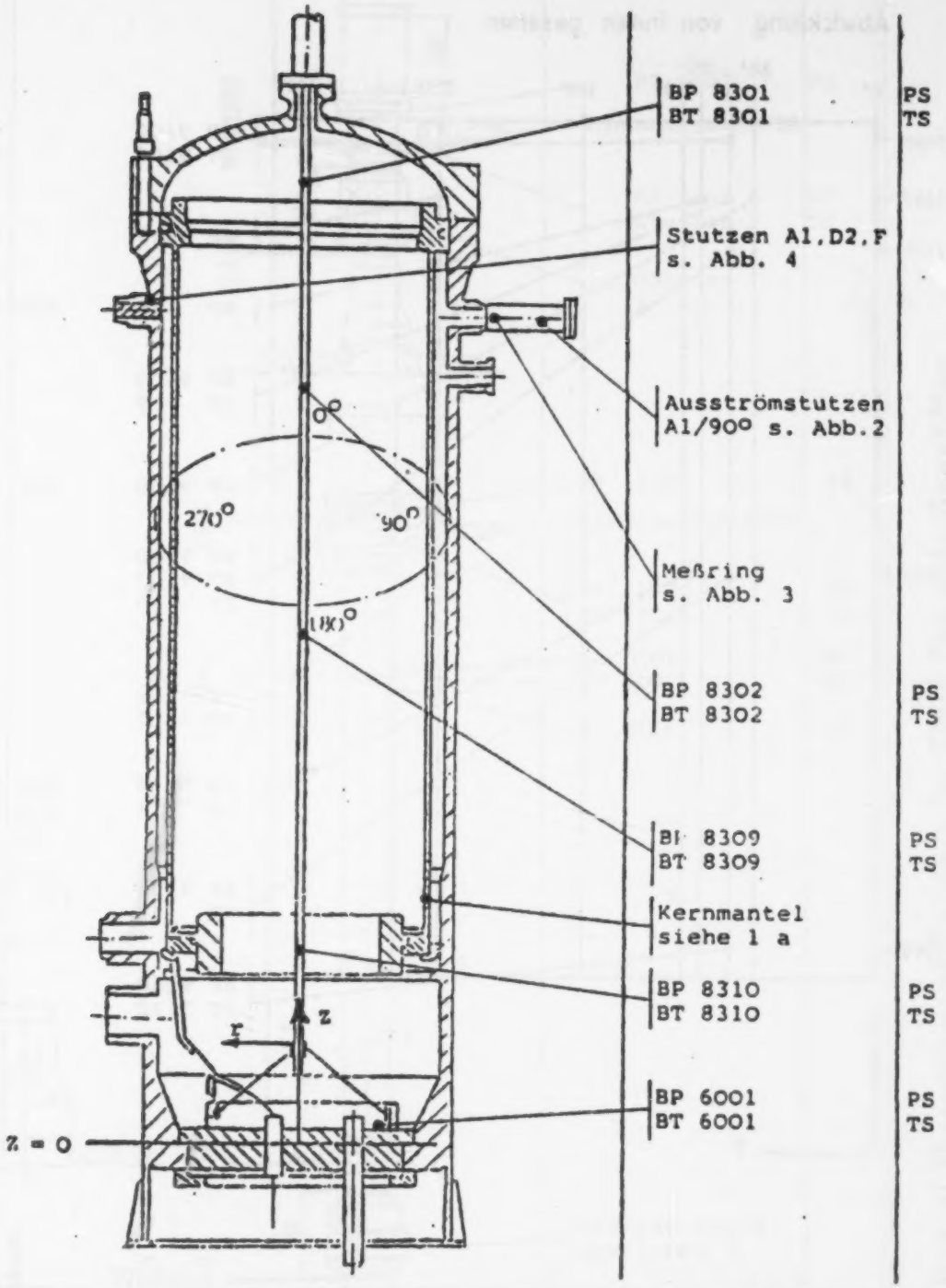
- Best-Estimate Codes
- Design Codes
- Licensing Codes

— DECISIONS FOR MAIN TEST SERIES

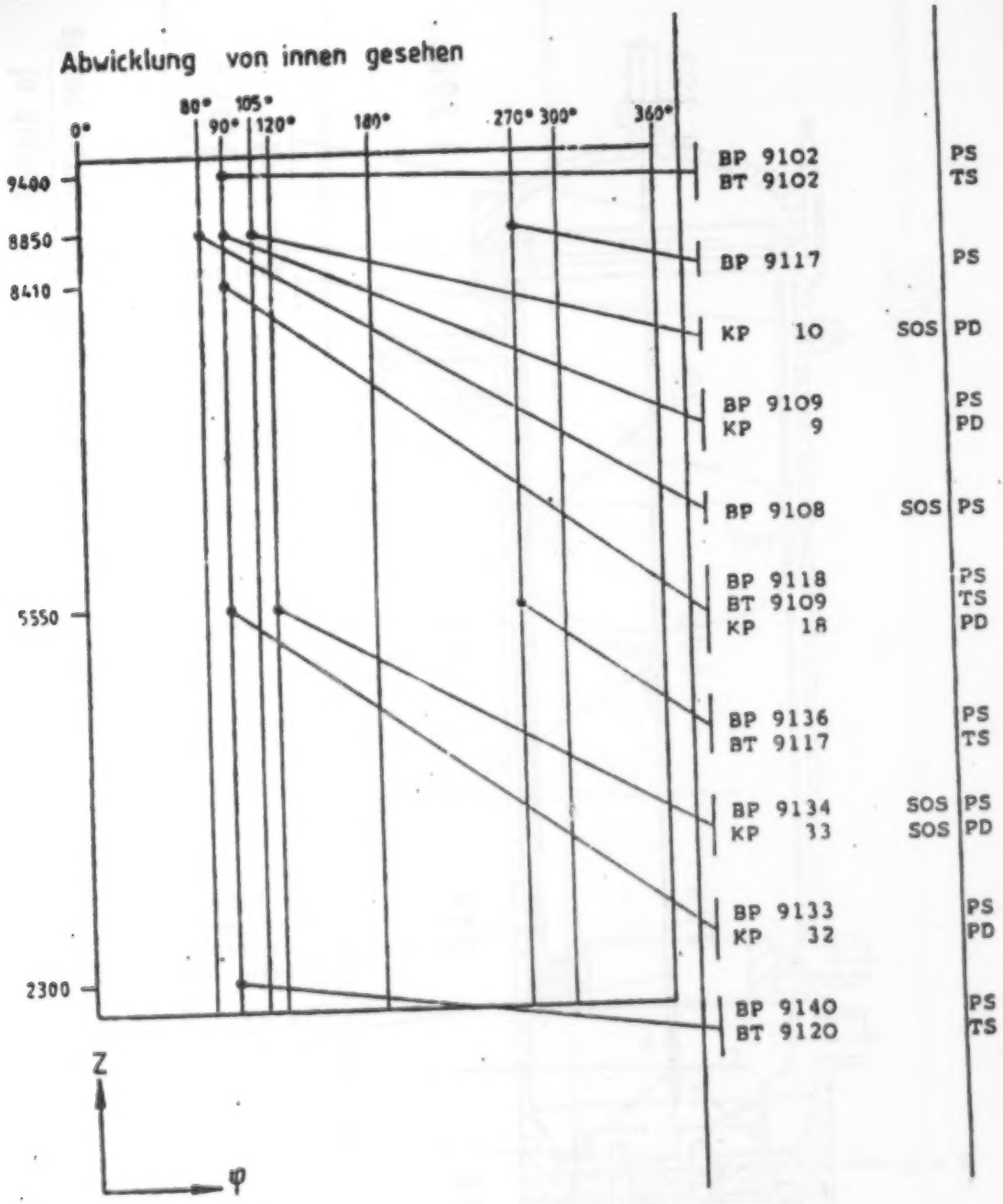
- Number of Tests
- Initial- and Boundary Conditions
- Types and Locations of Sensors

Test No	Presuure bar	Upper Core Temp. °C	Downcomer Tang °C	Subcooling Downcomer Degrees	Length of break nozzle m
V29.2	90	293	273	30	4.524
V31	110	308	268	50	1.369
V31.1	110	308	268	50	1.369

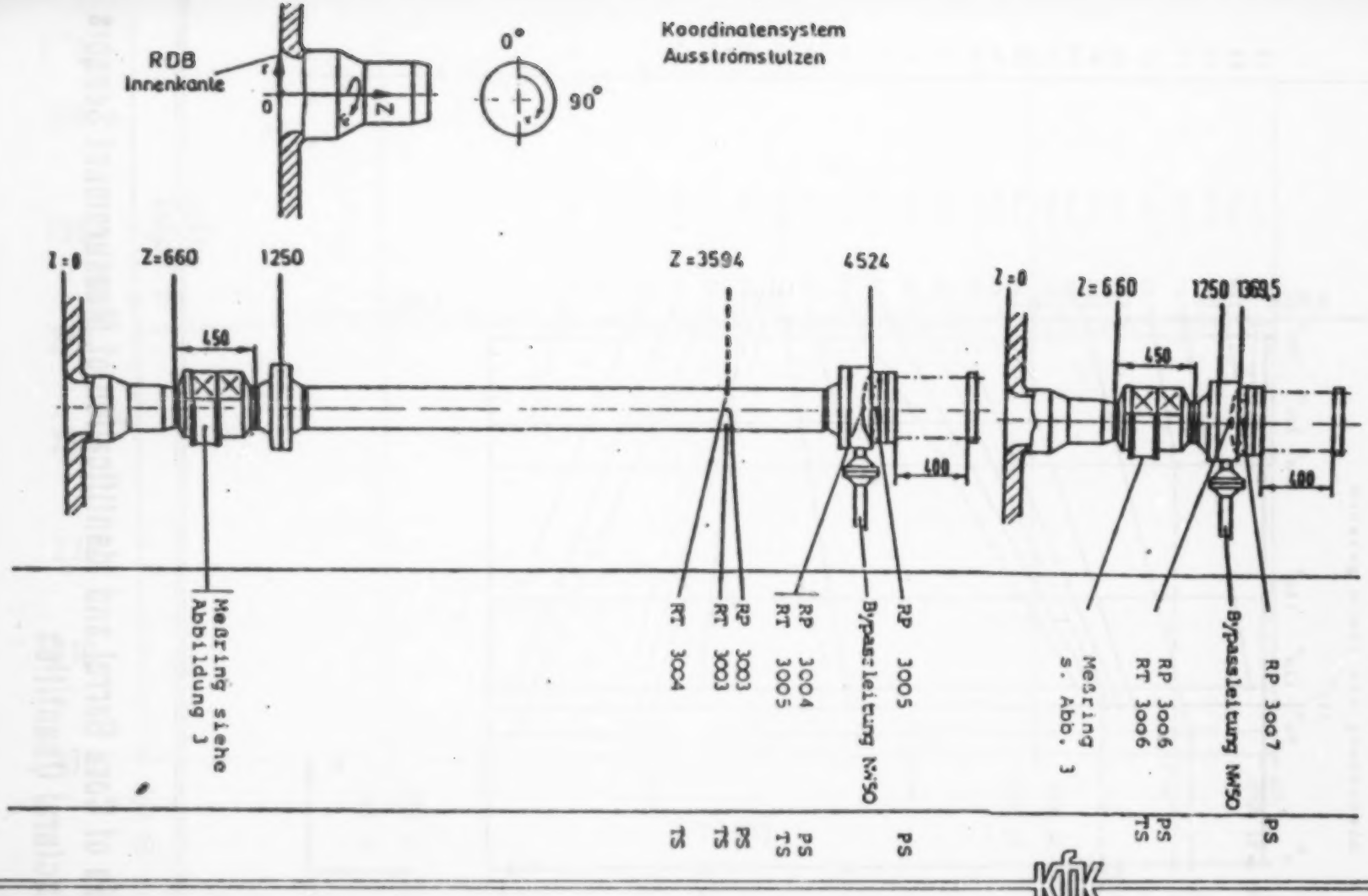
Test Matrix for Preliminary Blowdown Tests



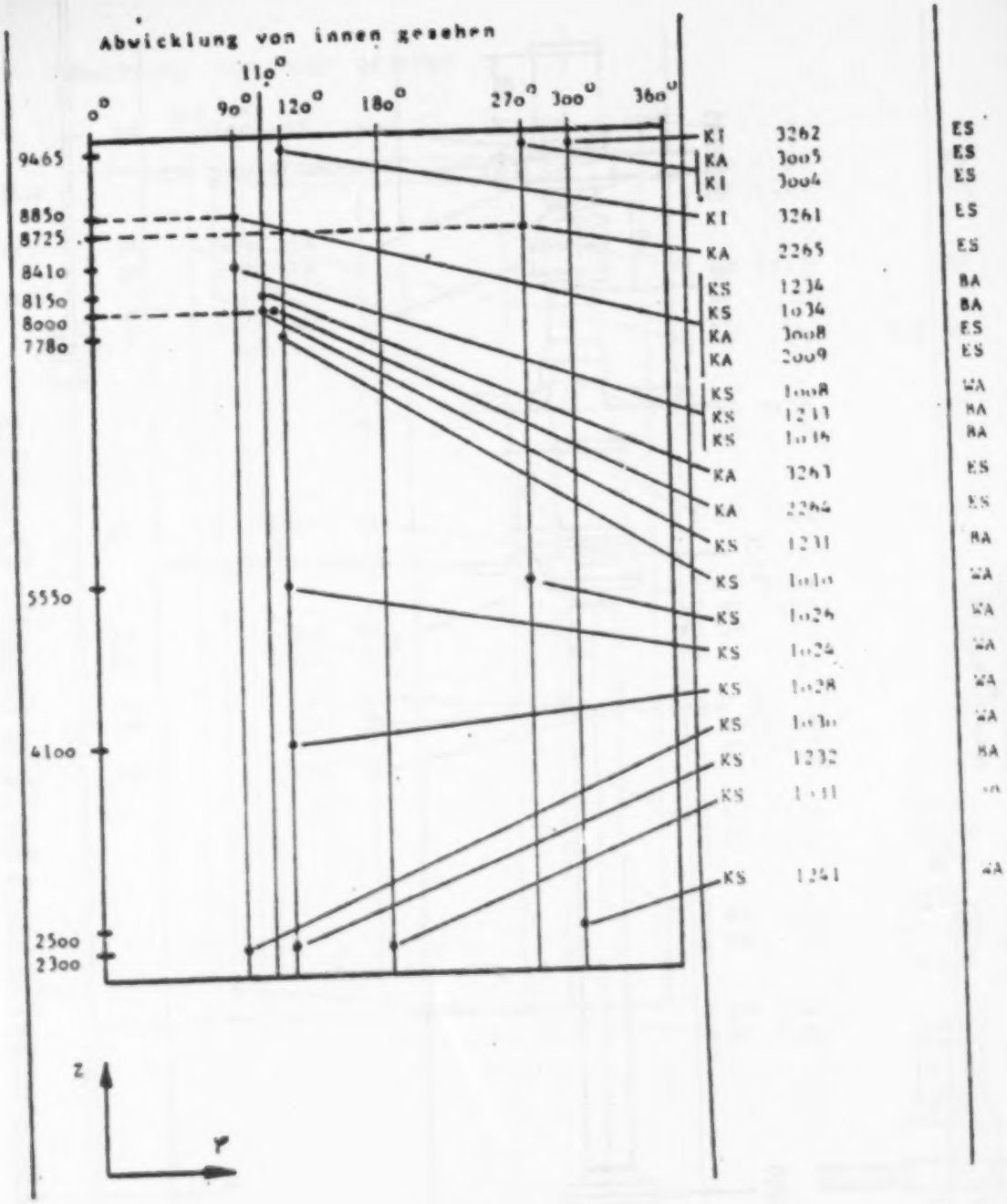
Identification of Measurement Sensors for Fluiddynamic Quantities Inside the Core Barrel



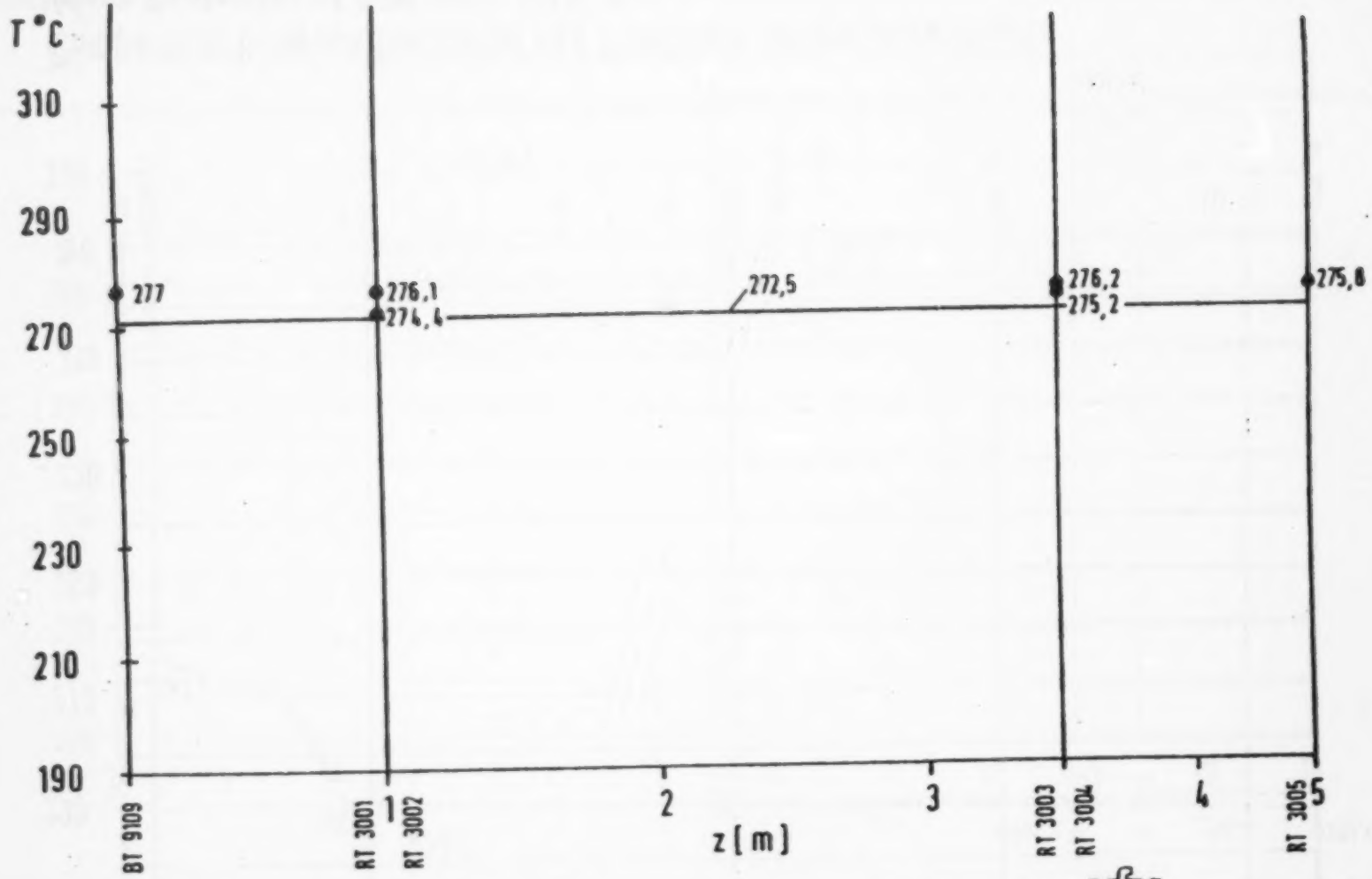
Wrap-Up of Core Barrel and Identification of Measurement Sensors for Fluiddynamic Quantities



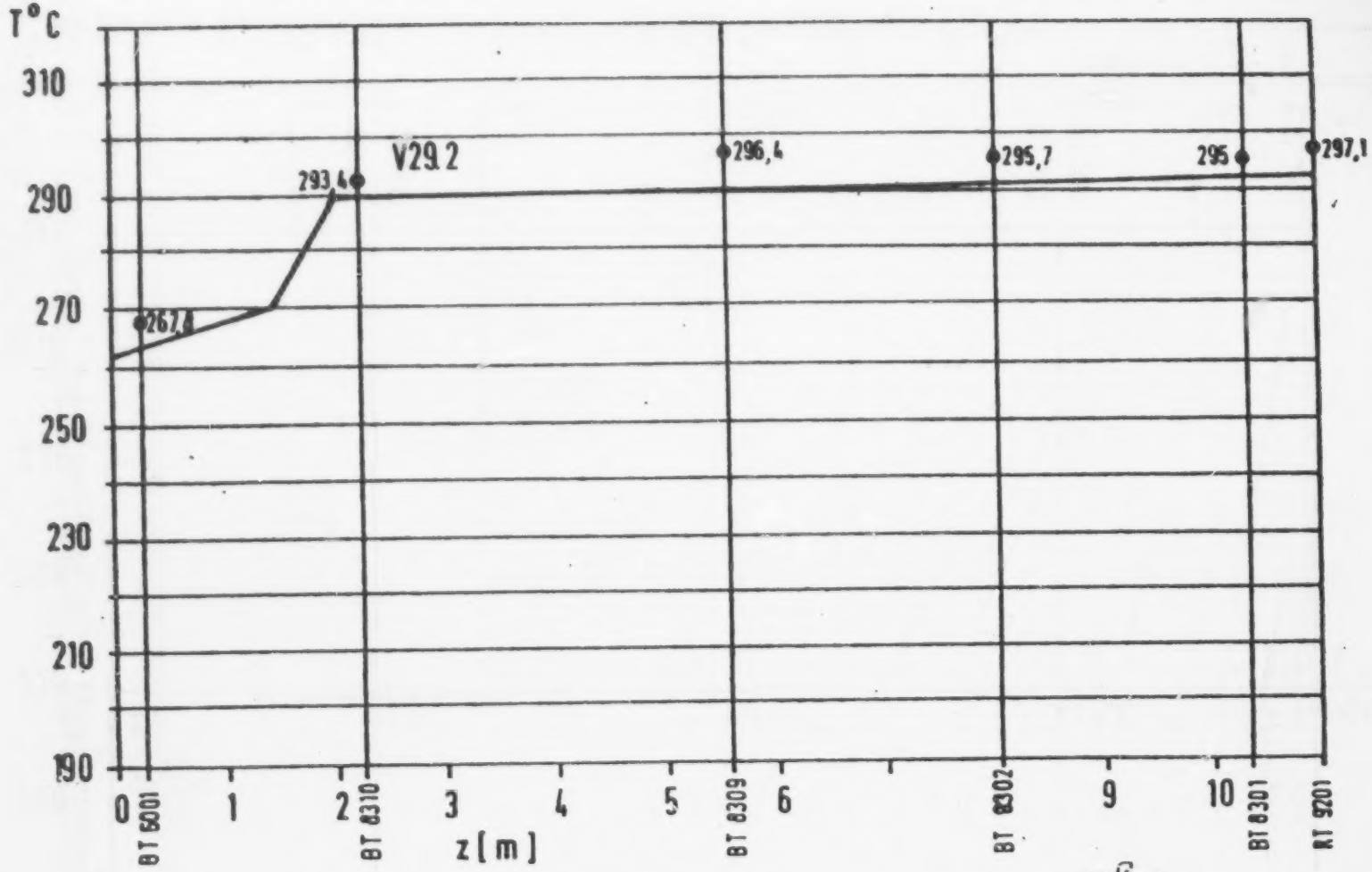
Identification of Measurement Sensors for Fluid Dynamic Quantities in the Long and Short Break Nozzles



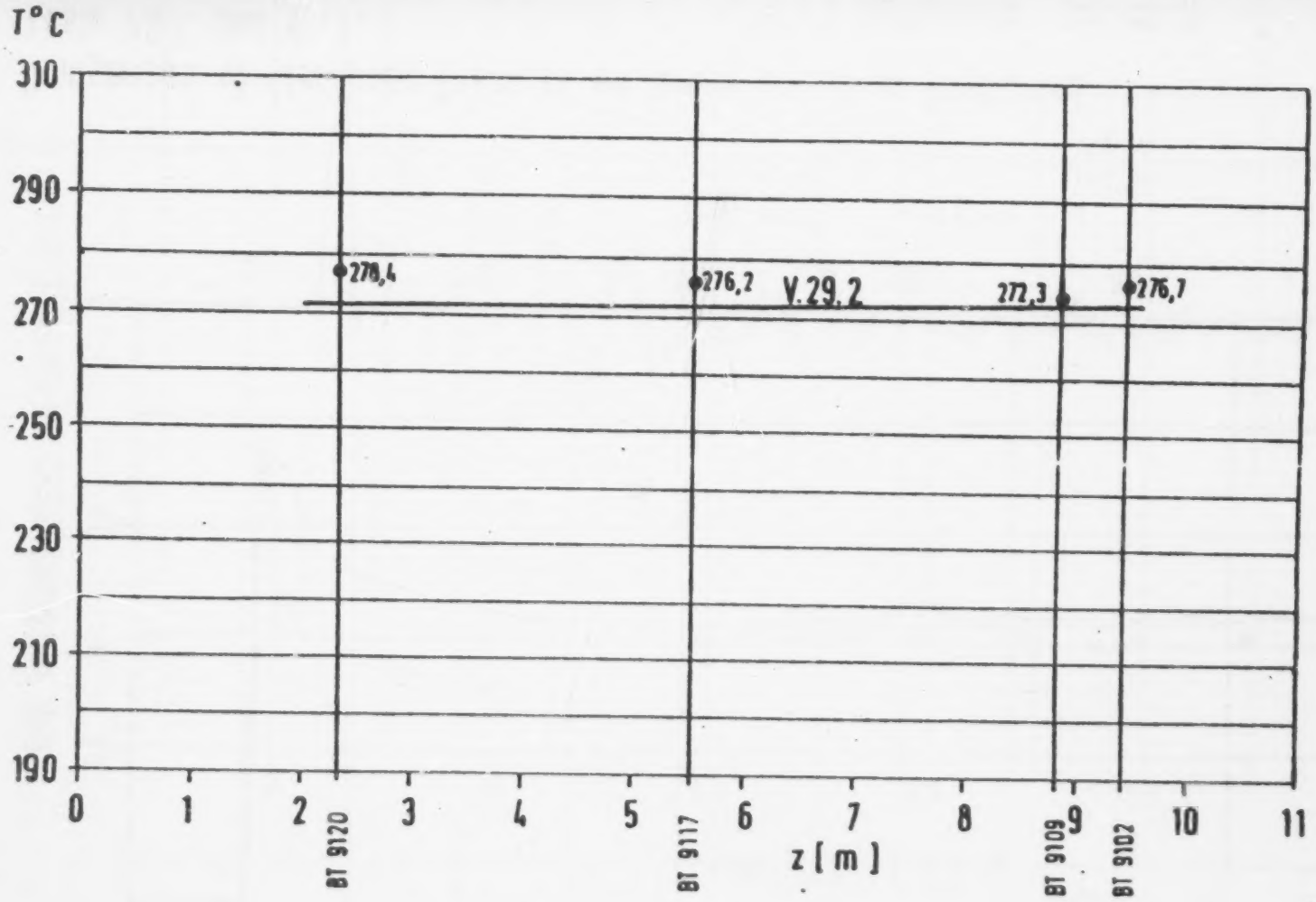
Wrap-Up of Core Barrel and Identification of Measurement Sensors for Structural Quantities



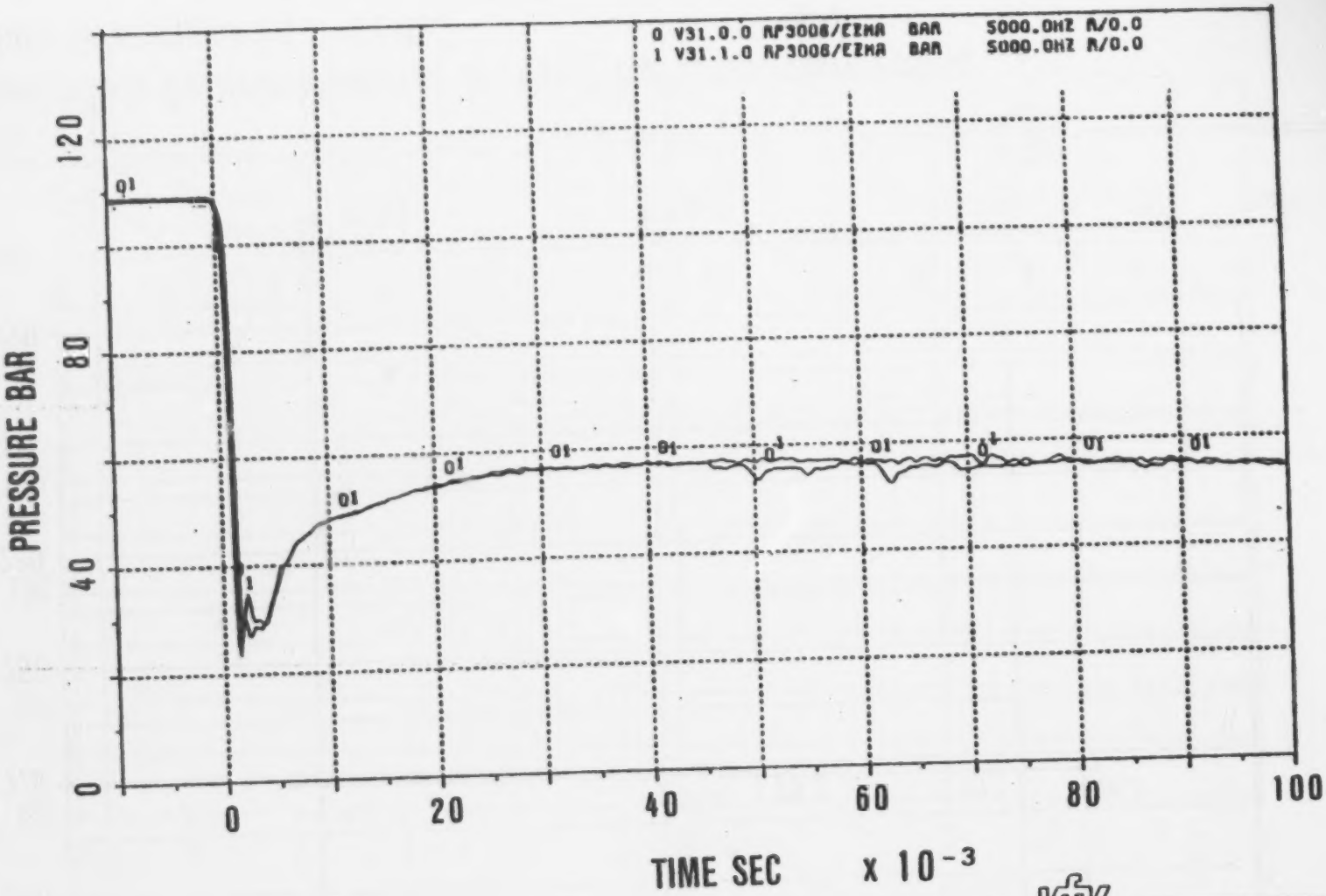
Comparison Between Measured and Predicted Temperature Profiles Along Long Break Nozzle for V29.2



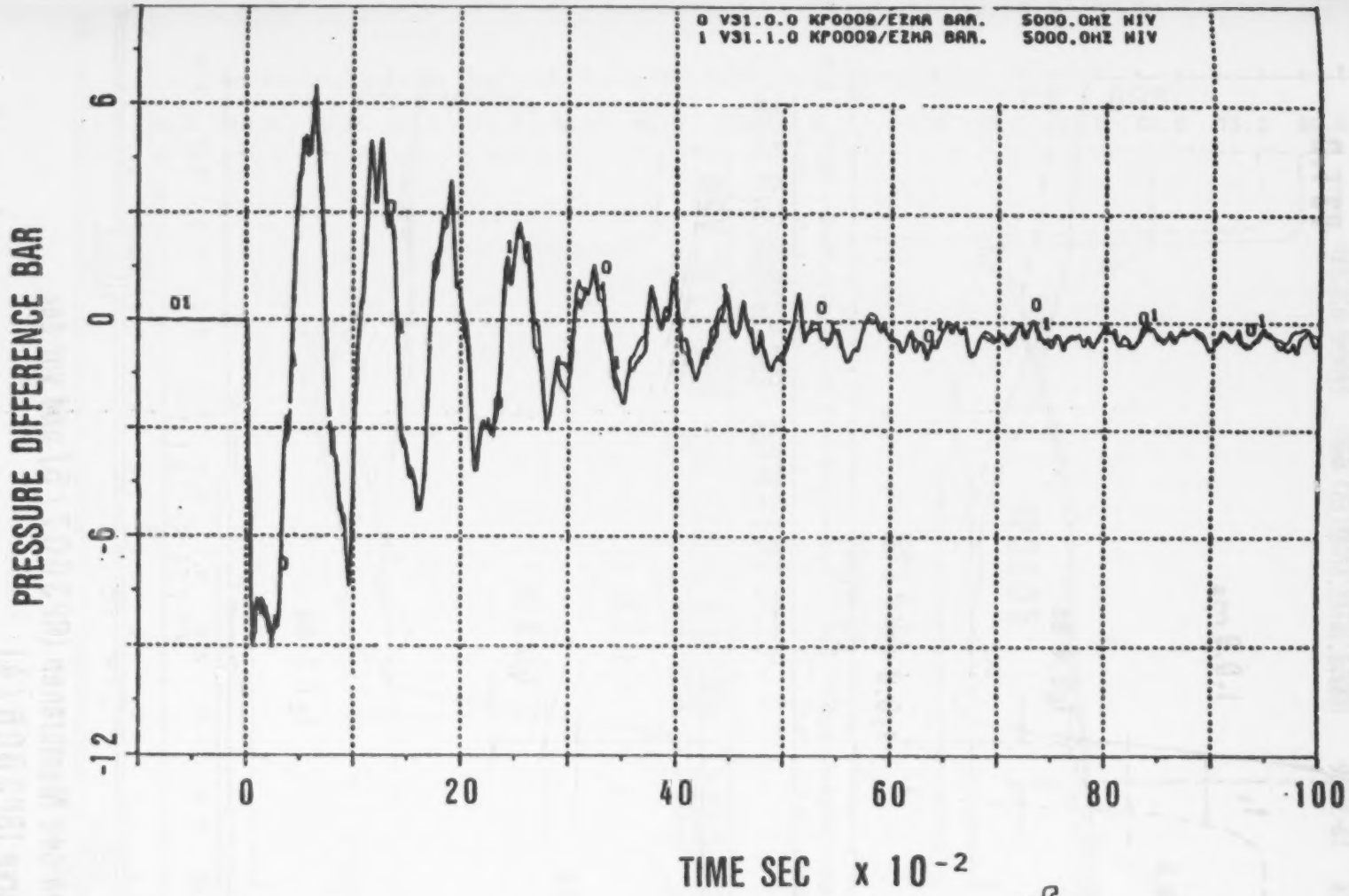
Comparison Between Measured and Predicted Temperature Profile Along Centerline of RPV for V29.2



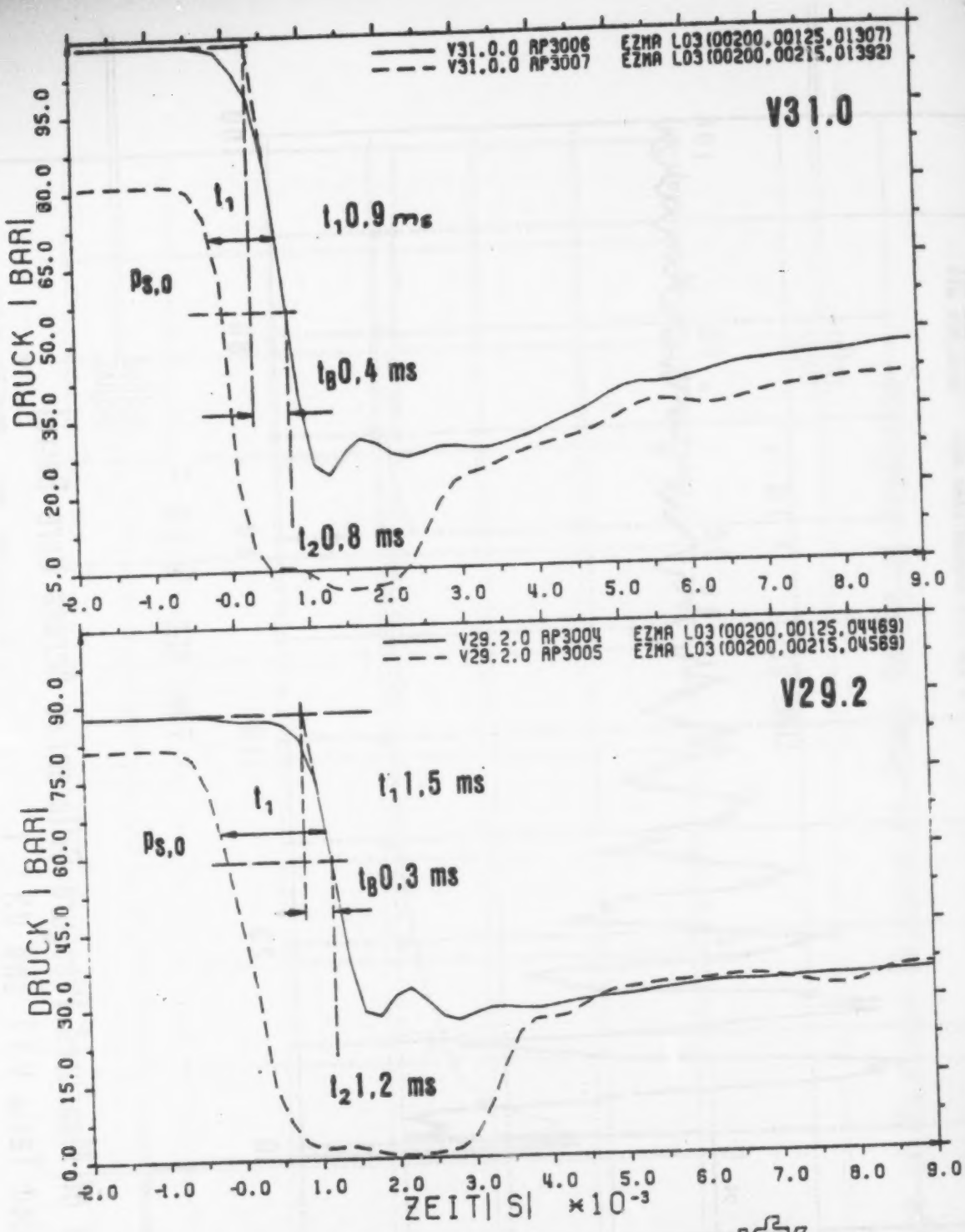
Comparison Between Measured and Predicted Temperature Profils Along Downcomer for V29.2



Comparison of Measured Pressure on Break Nozzle of Replicated Tests V31 and V31.1



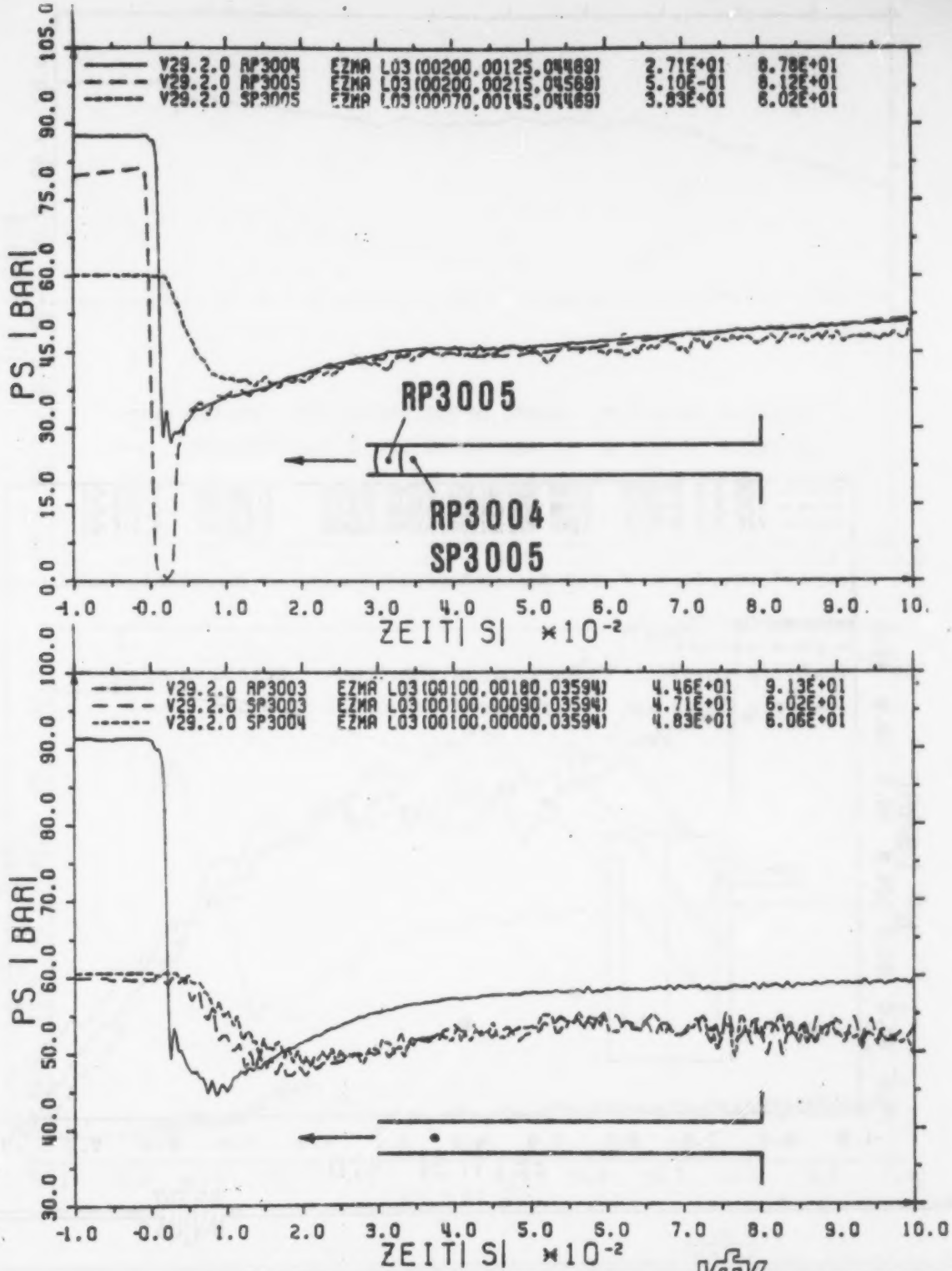
Comparison of Measured Pressure Differences Across Core Barrel for Replicated Tests V31 and V31.1



Druck zwischen den Membranen (RP3007 / 5) und vor der inneren Membran (RP3006 / 4)

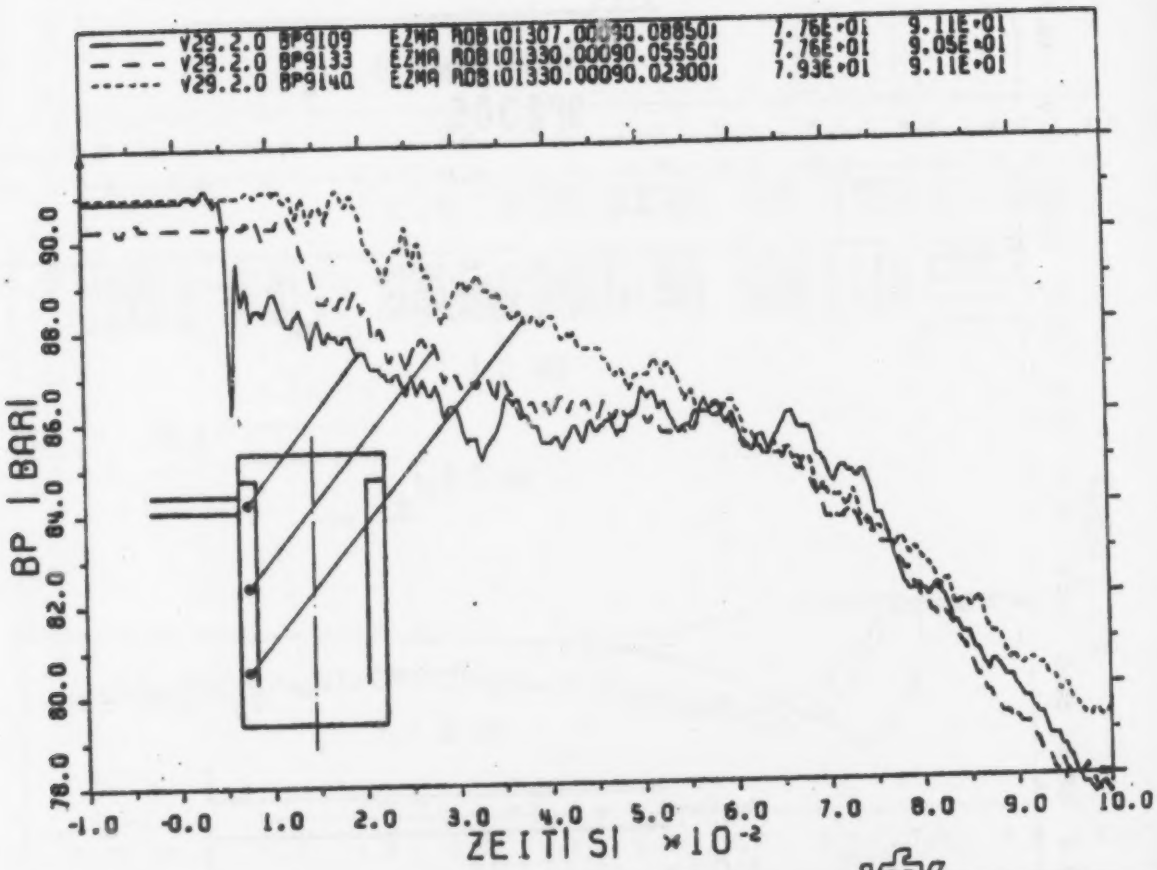
Anlage 6

Pressure Between Membranes (RP3007 / 5) and in Front of the Inside Membranes (RP3006 / 4)



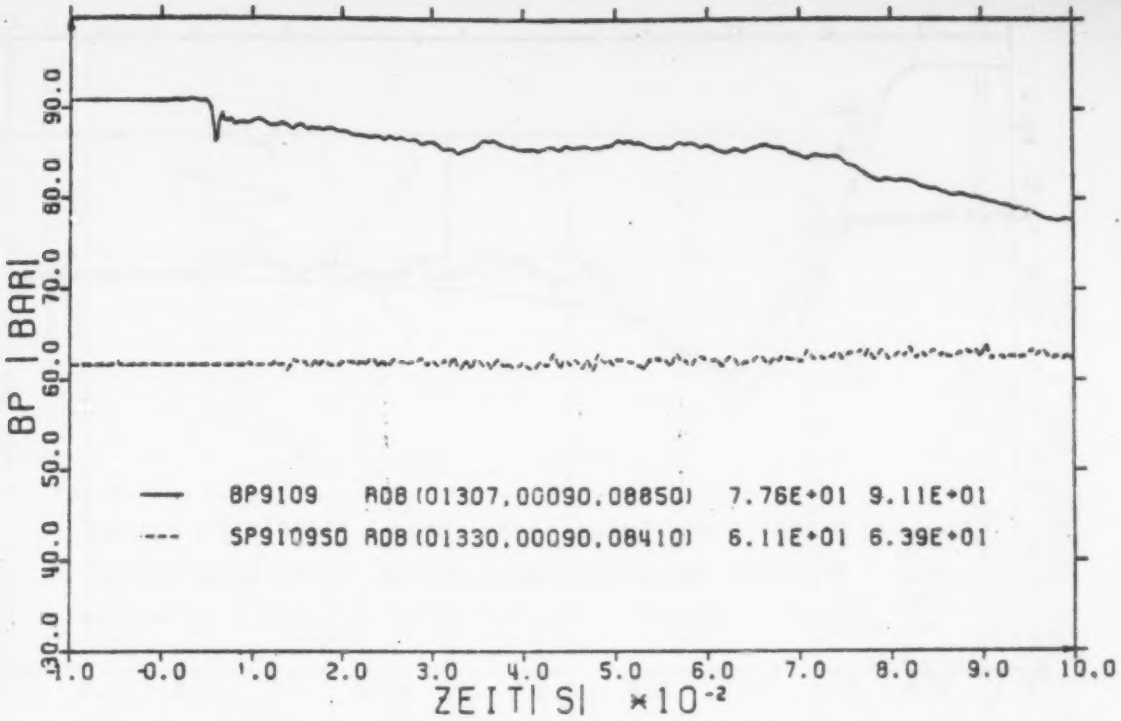
V29.2 / Druck (RP) und Sattdampfdruck (SP) im Stutzen
Pressure (RP) and Saturation Pressure (SP) in the Nozzle

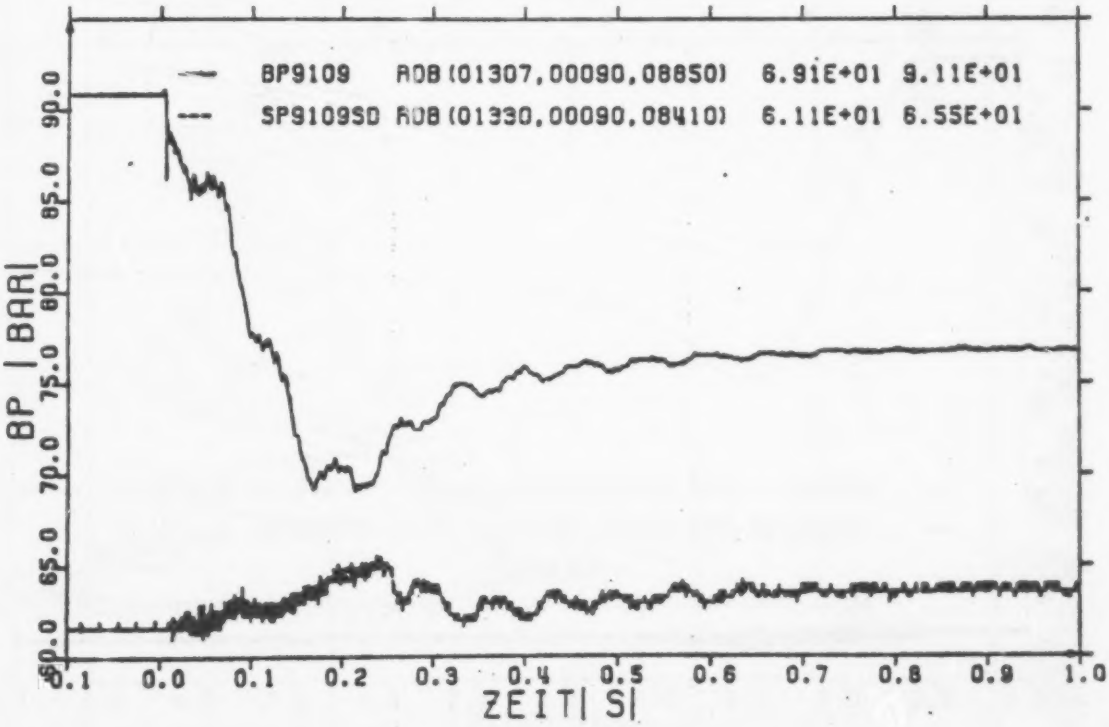
Anlage 7



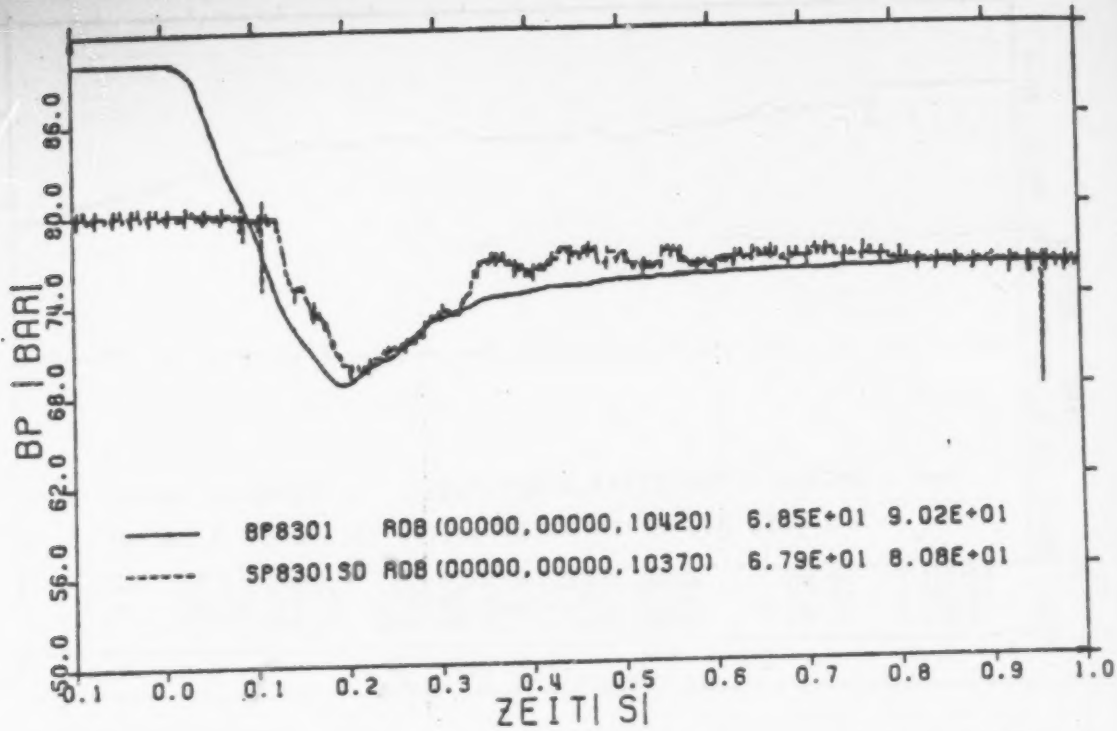
V29.2 / Druckwellen im Ringraum
Pressure Waves in Downcomer

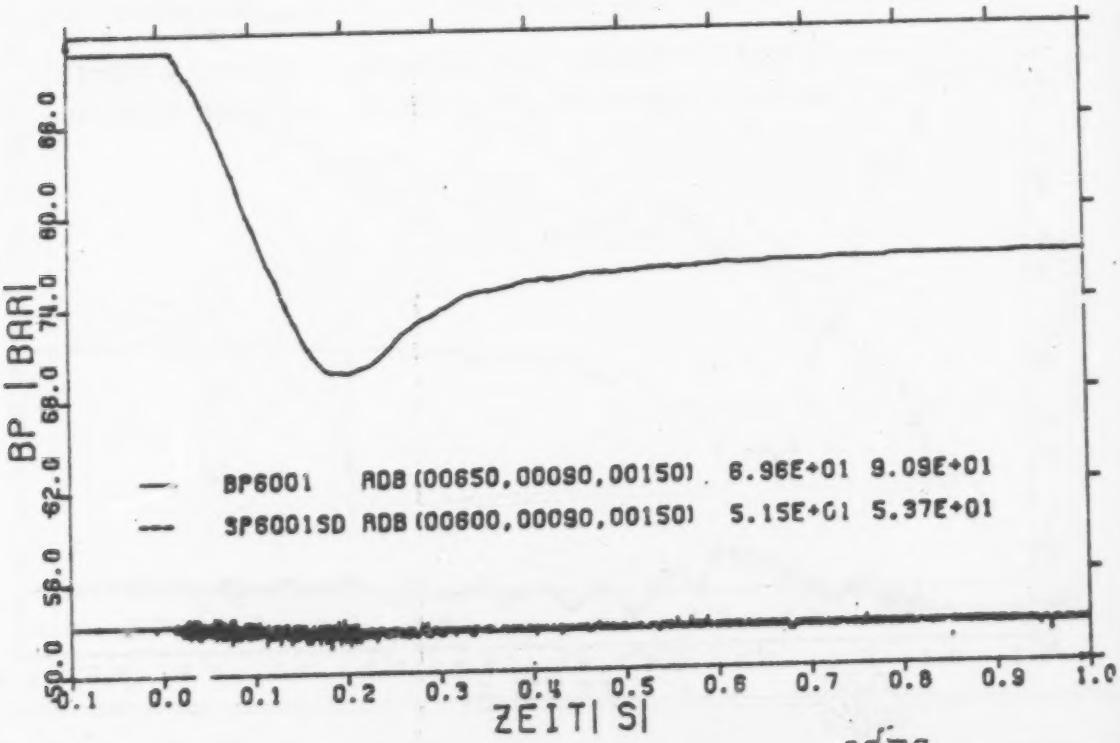
Anlage 11



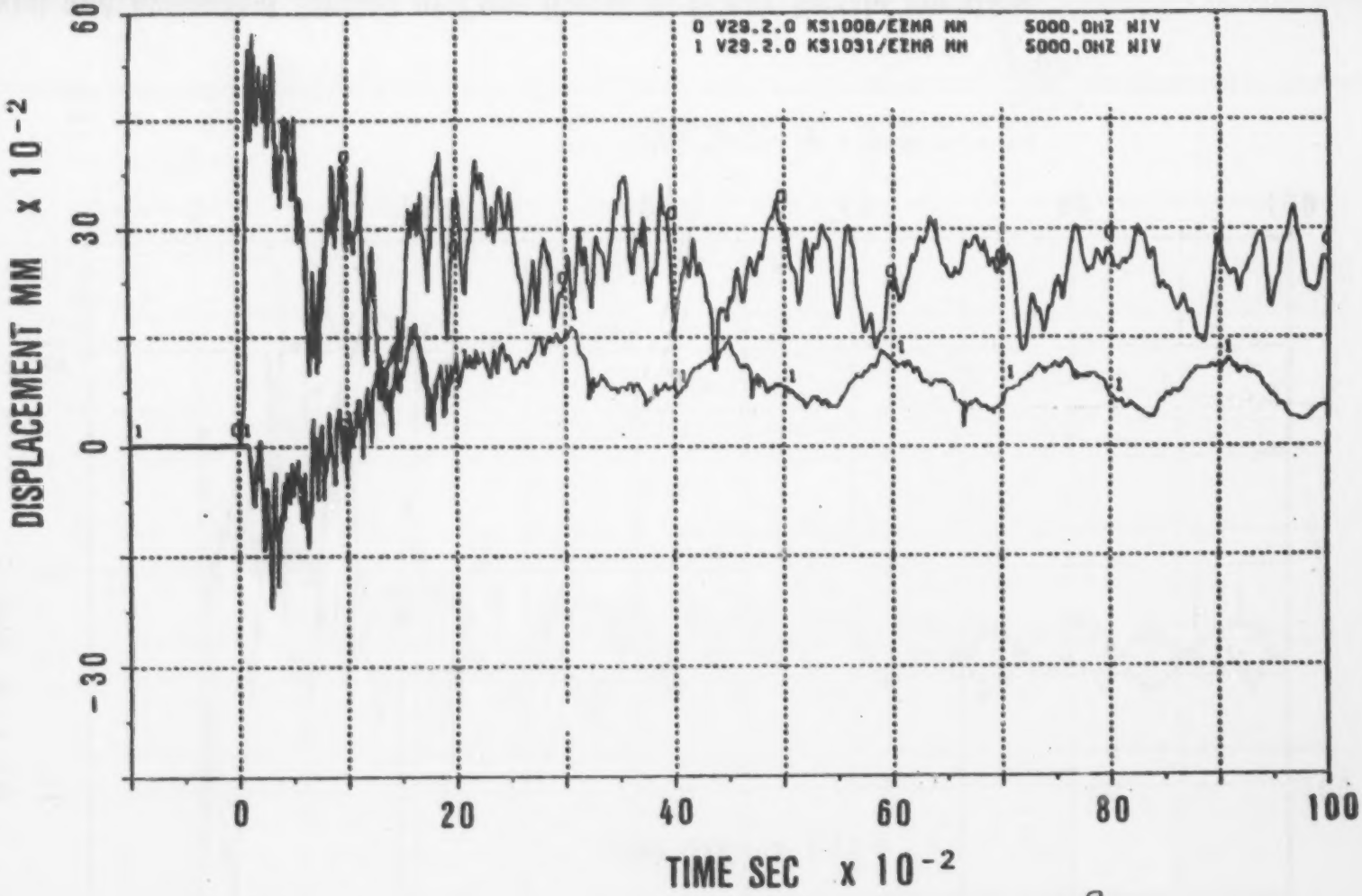


Pressure and Saturation Pressure in Downcomer at Break Nozzle Location over Different Times for V29.2

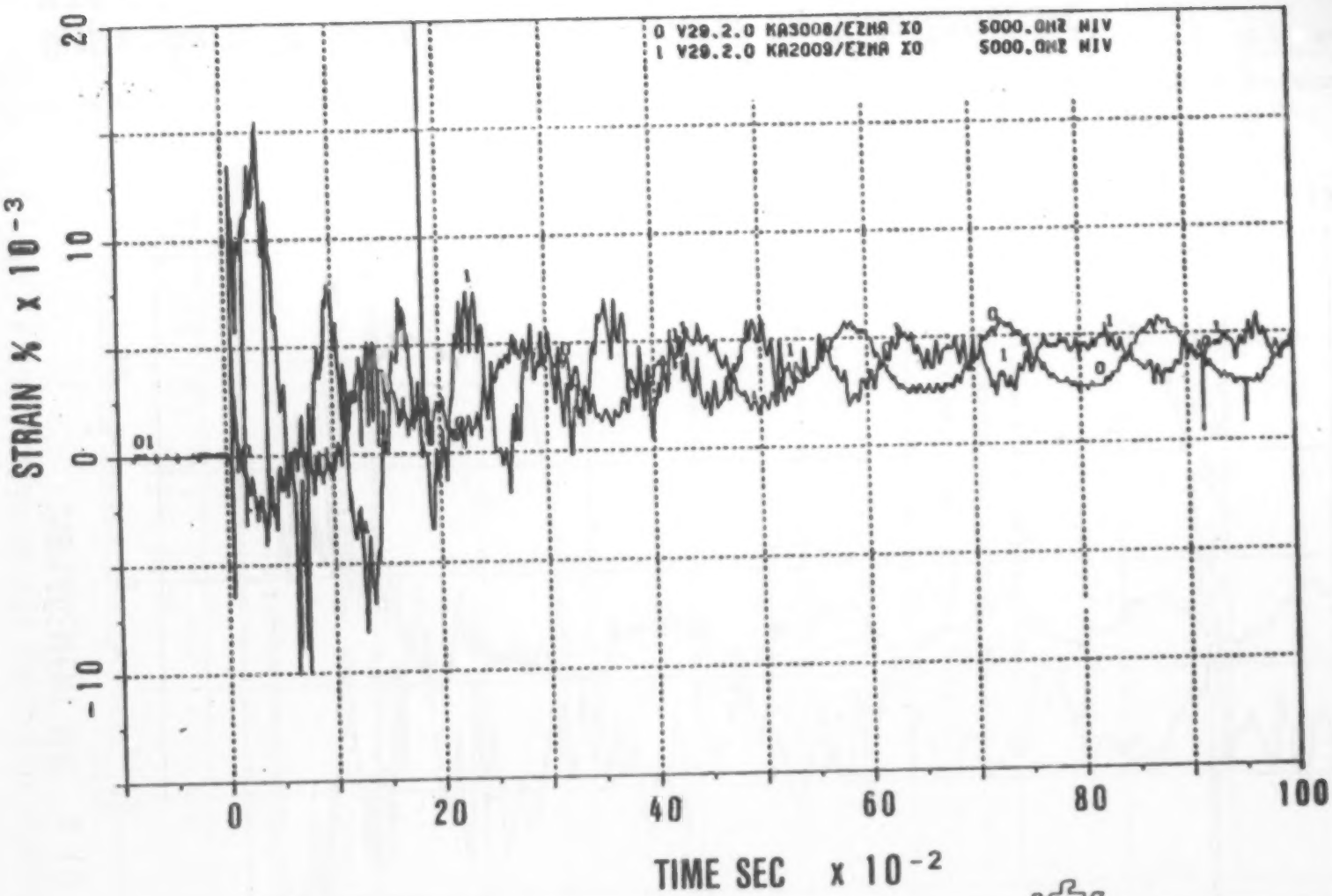




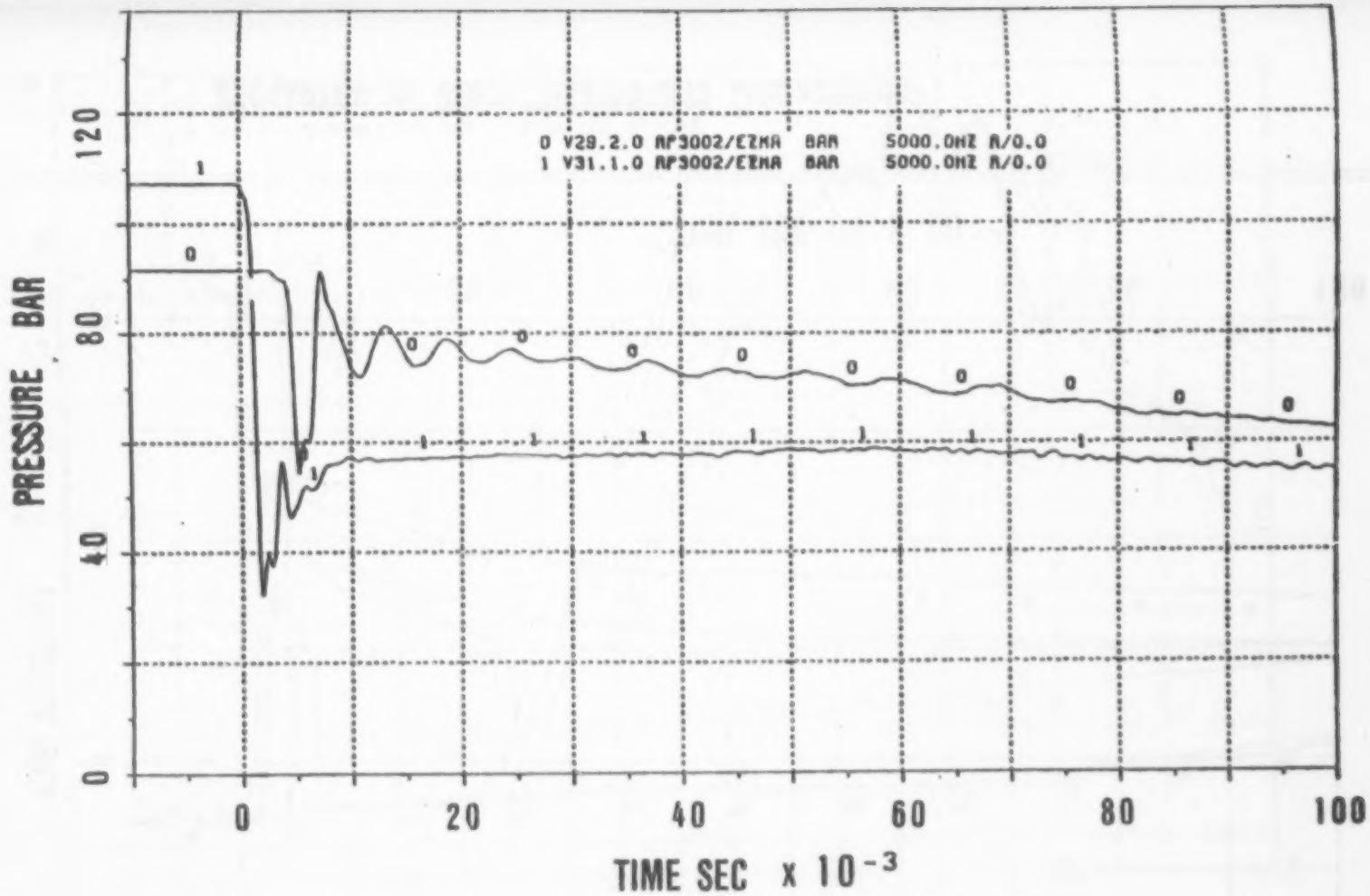
Upper and Lower Plena Pressures and Saturation Pressures for V29.2



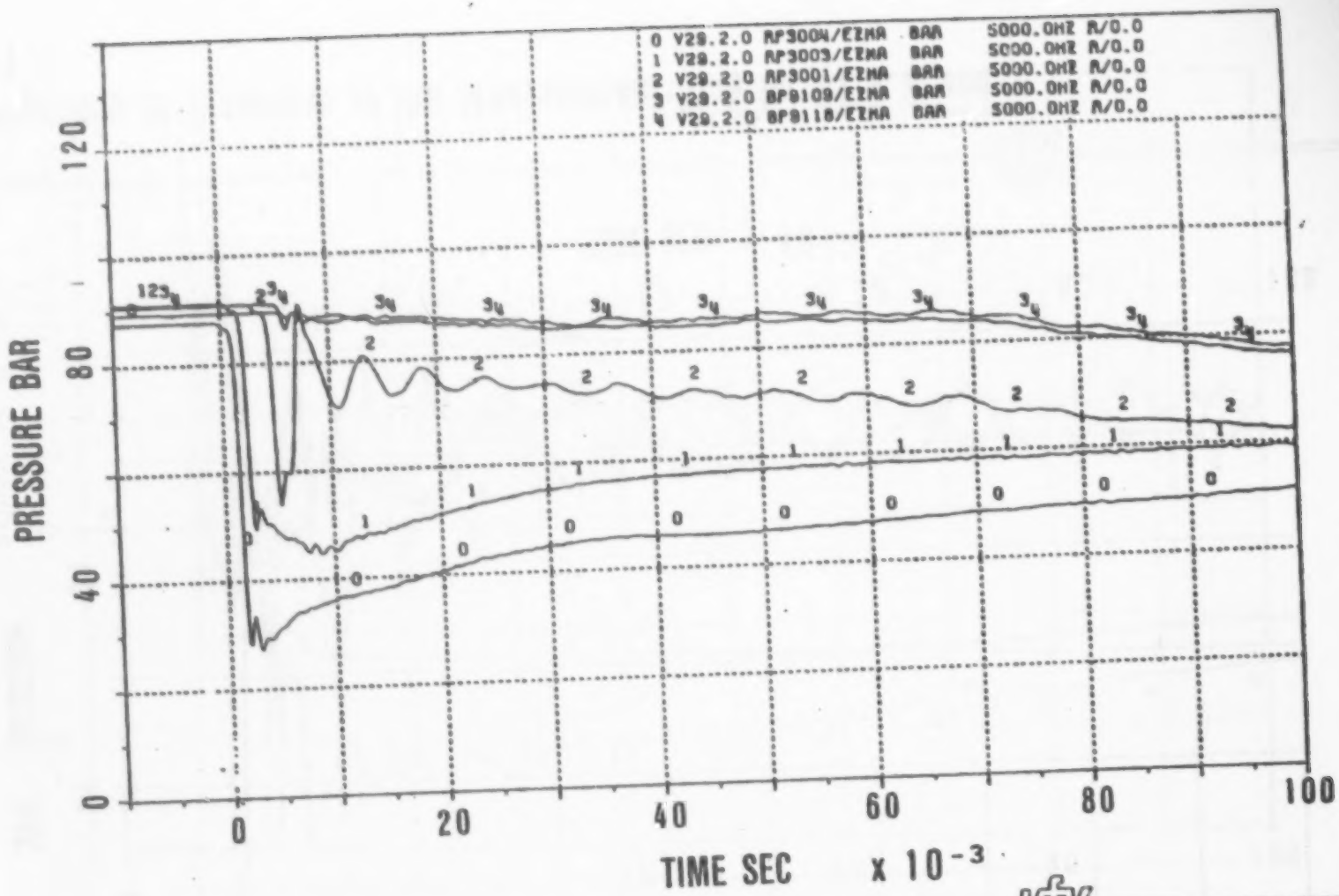
Displacements at Upper and Lower Parts of the Core Barrel for V29.2



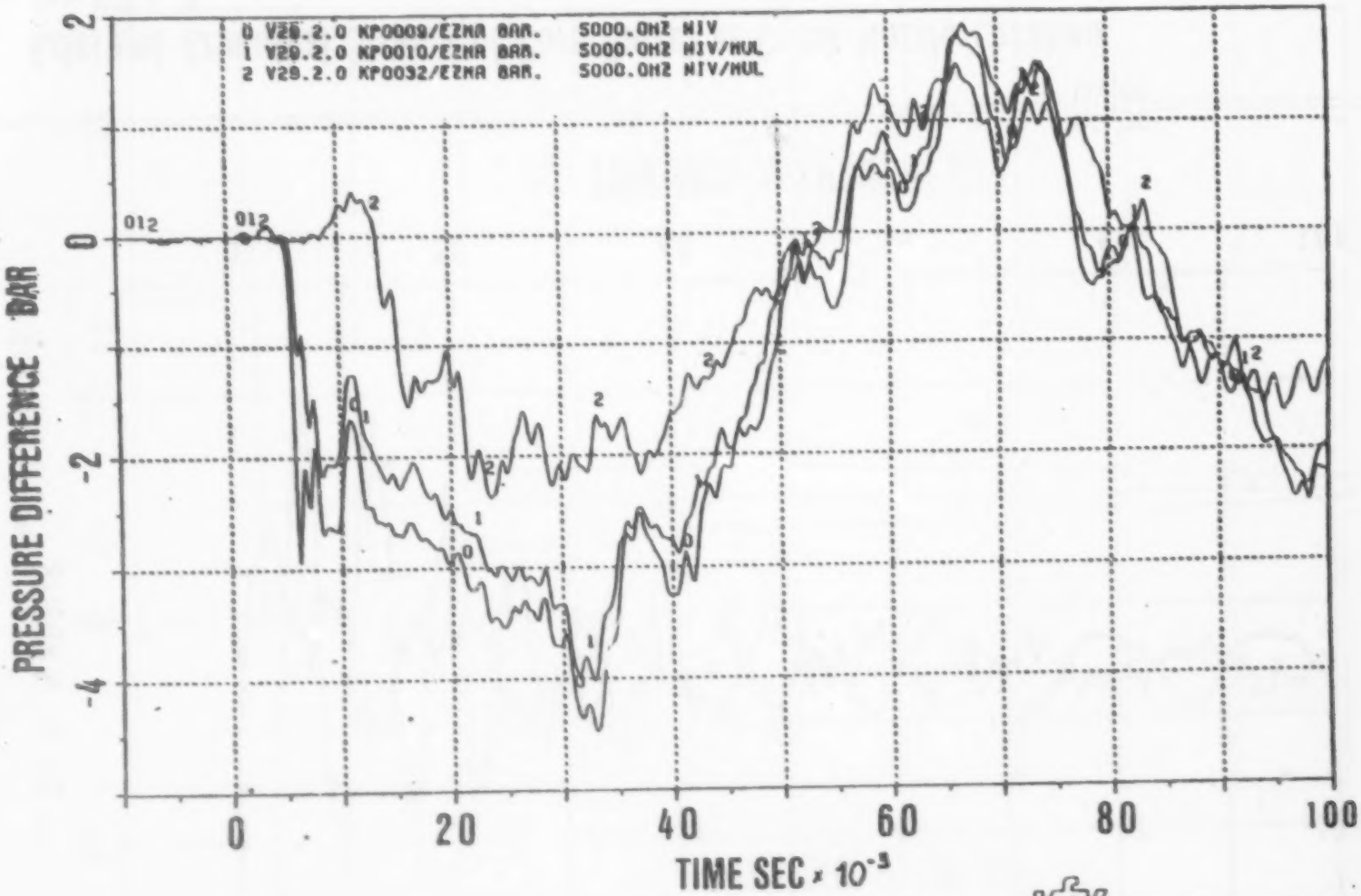
Axial and Azimuthal Strains of Core Barrel at Break Nozzle Location



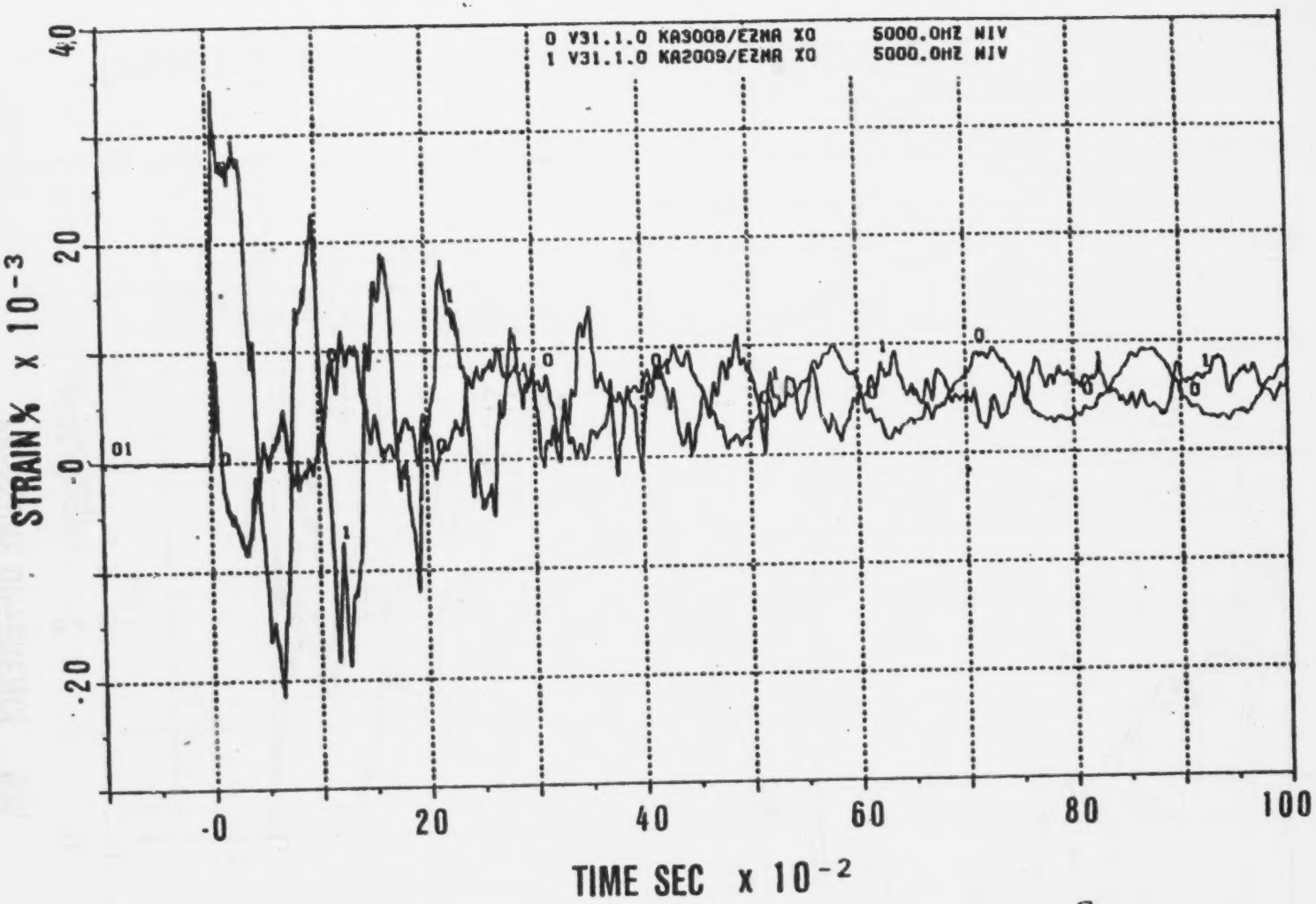
Comparison of Pressures at the Measurement Ring for V29.2 and V31.1



Pressures in Break Nozzle and Downcomer

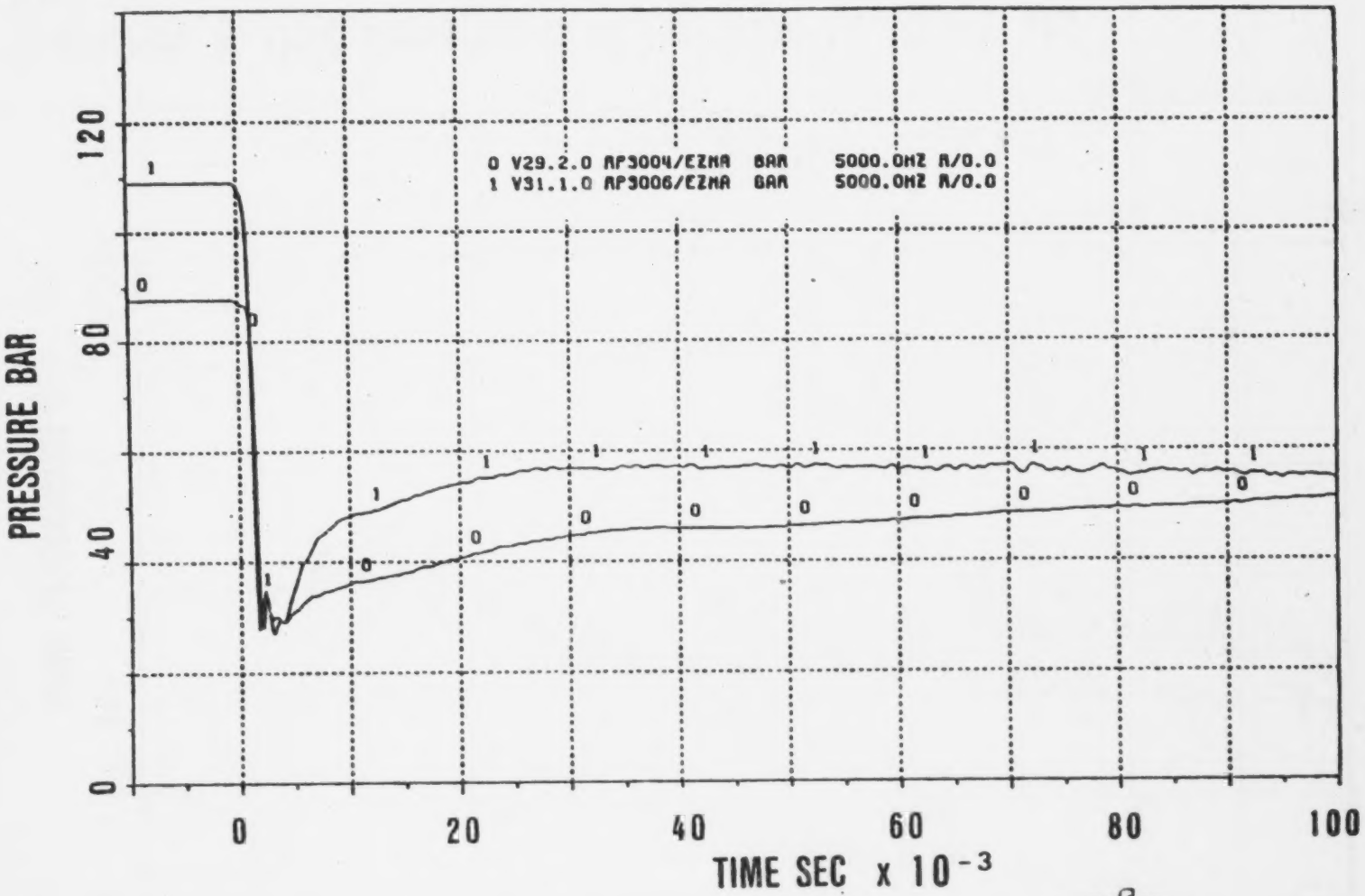


Pressure Differences Across Core Barrel for V29.2

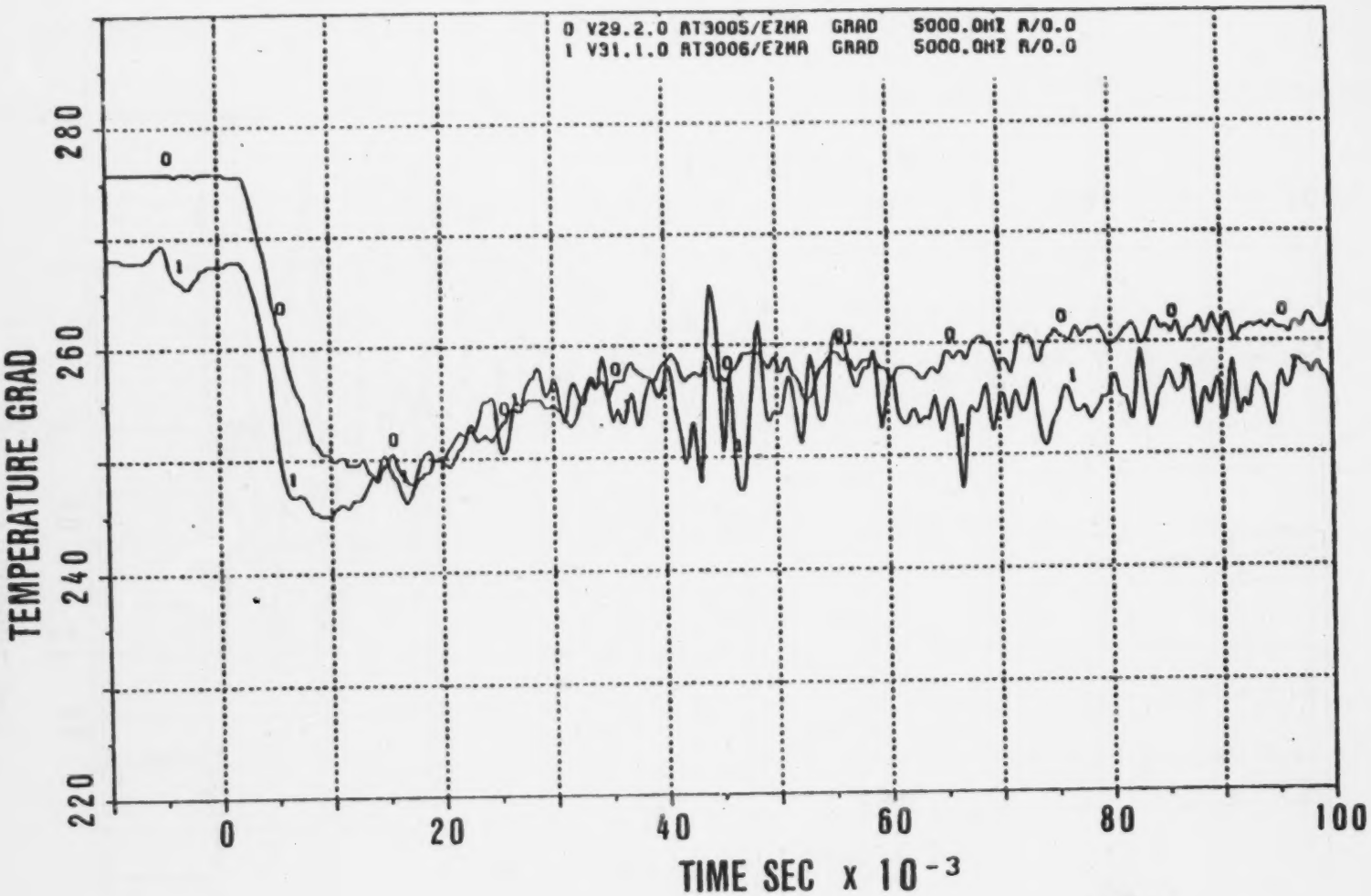


Axial and Azimuthal Strains of Core Barrel at Break Nozzle Location for V31.1

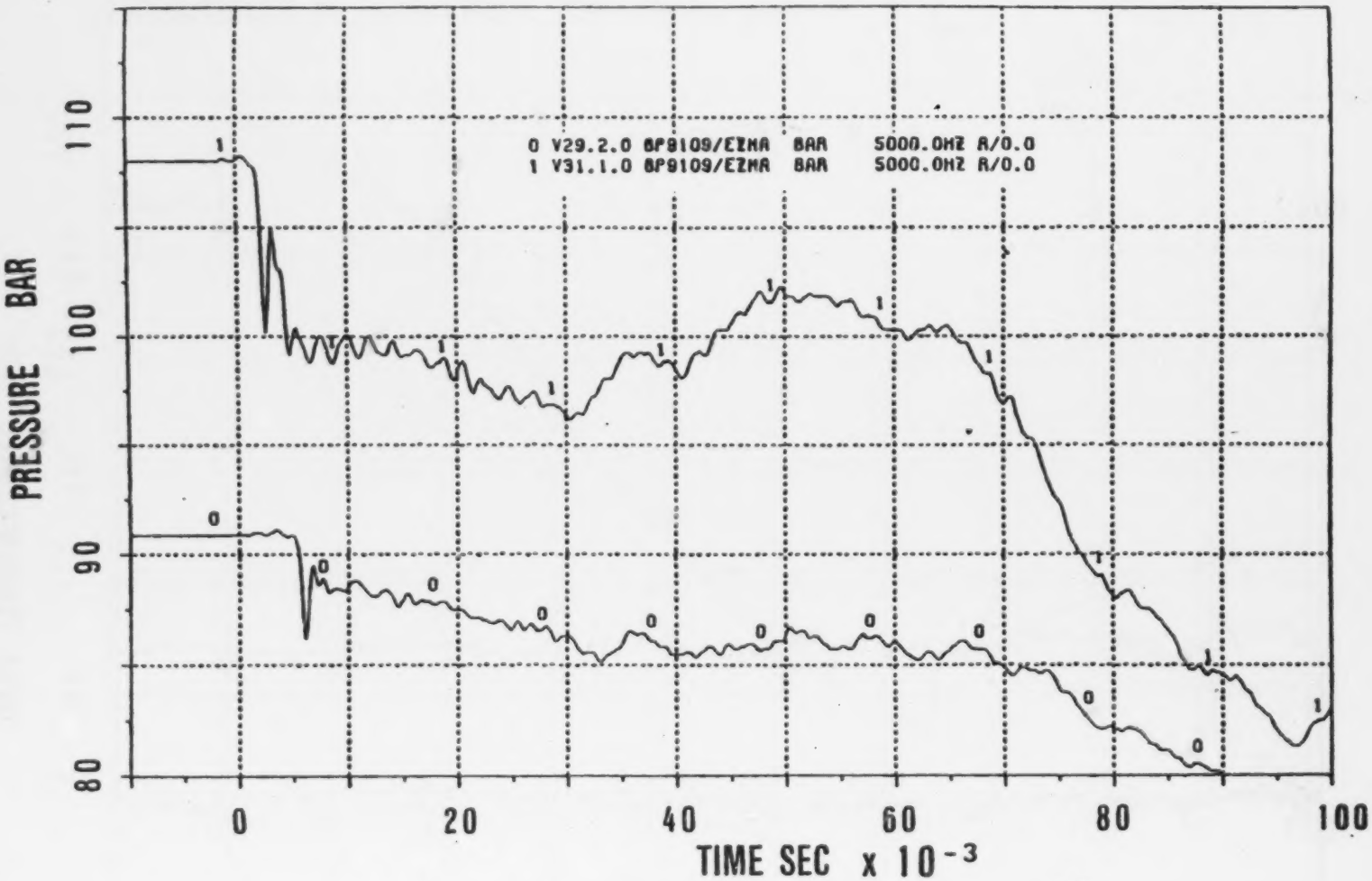
READER FRIENDLY
best copy available
pages tightly bound



Comparison of Pressures at the Break for V29.2 and V31.1

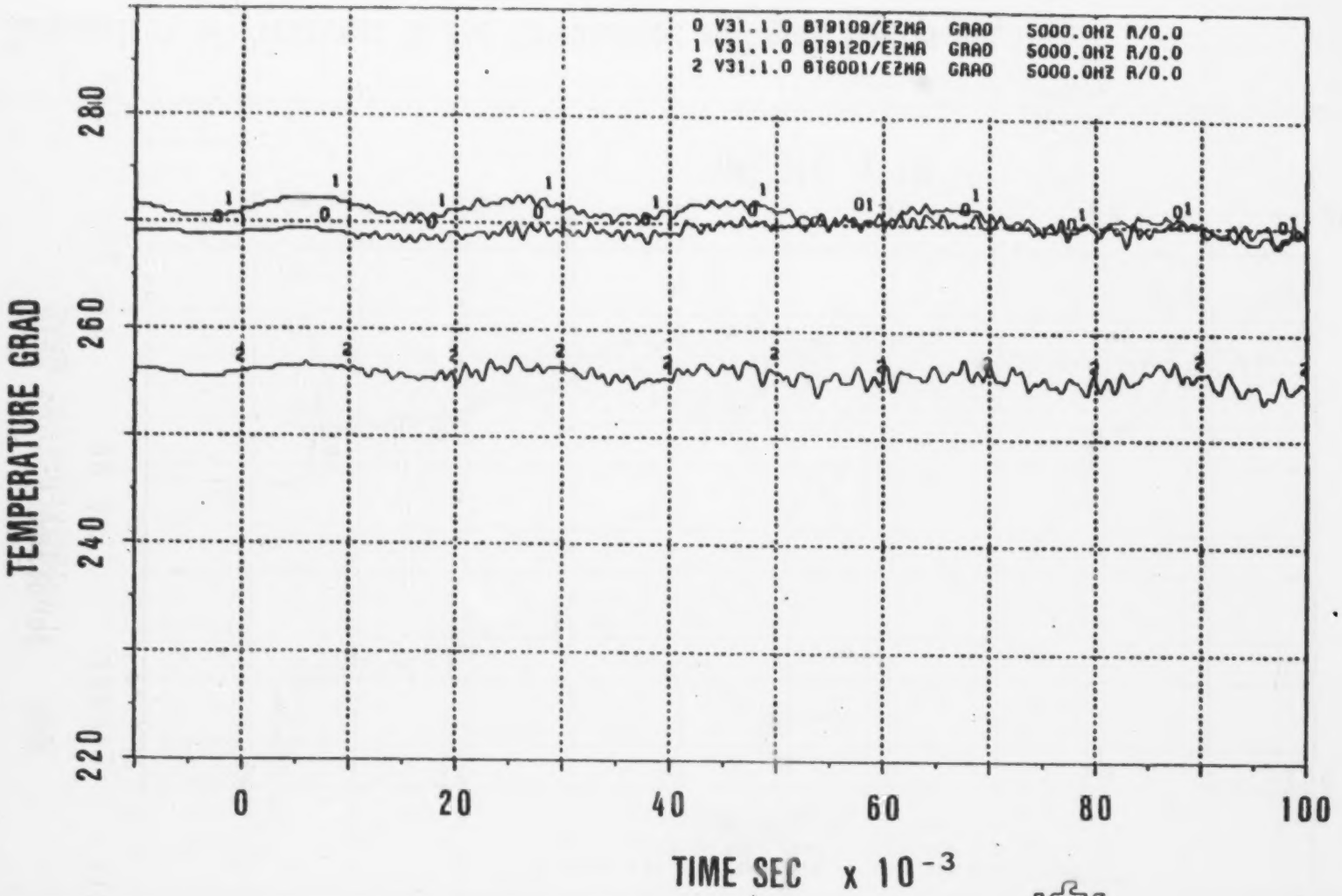


Comparison of Fluid Temperatures at the Break for V29.2 and V31.1

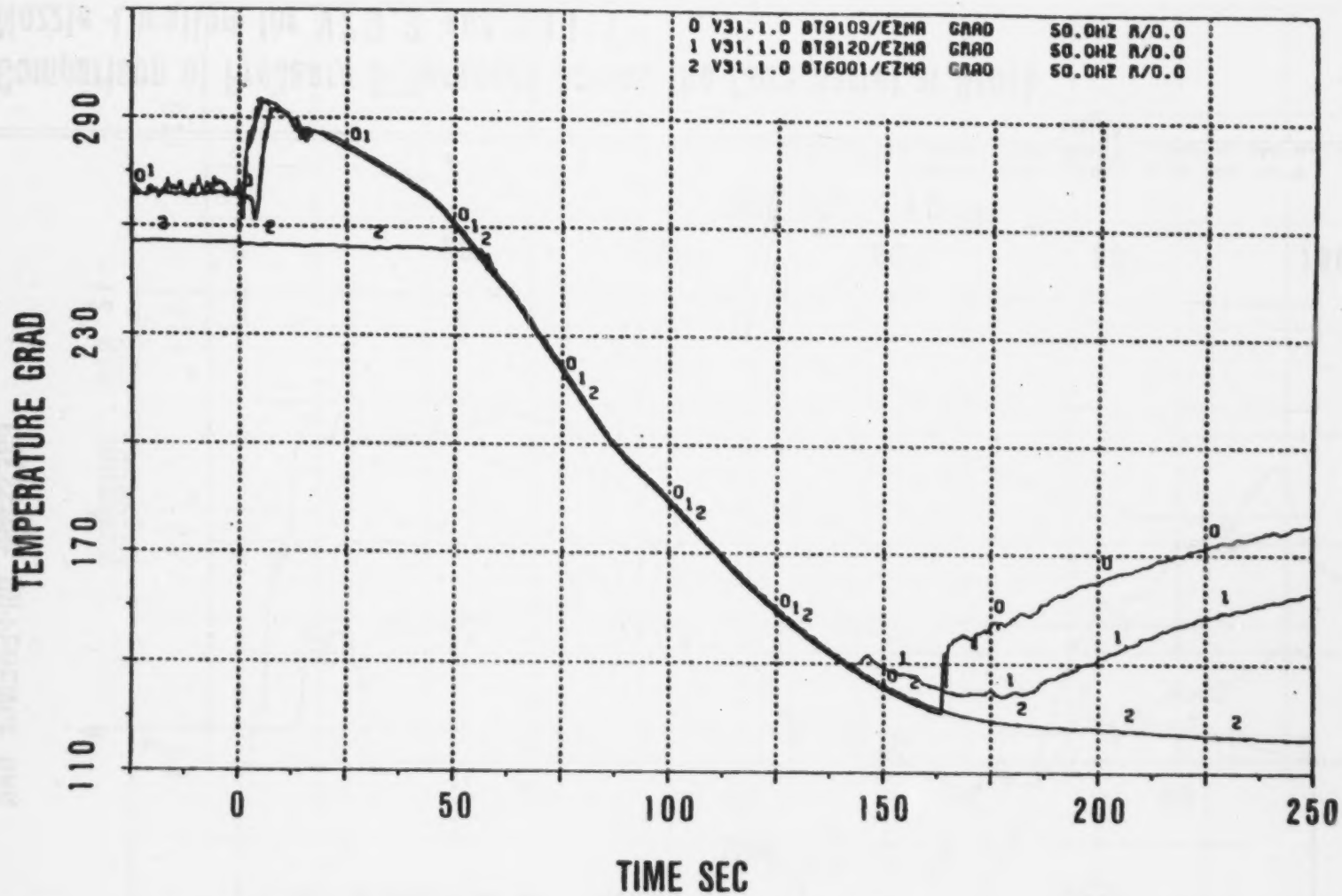


KTK

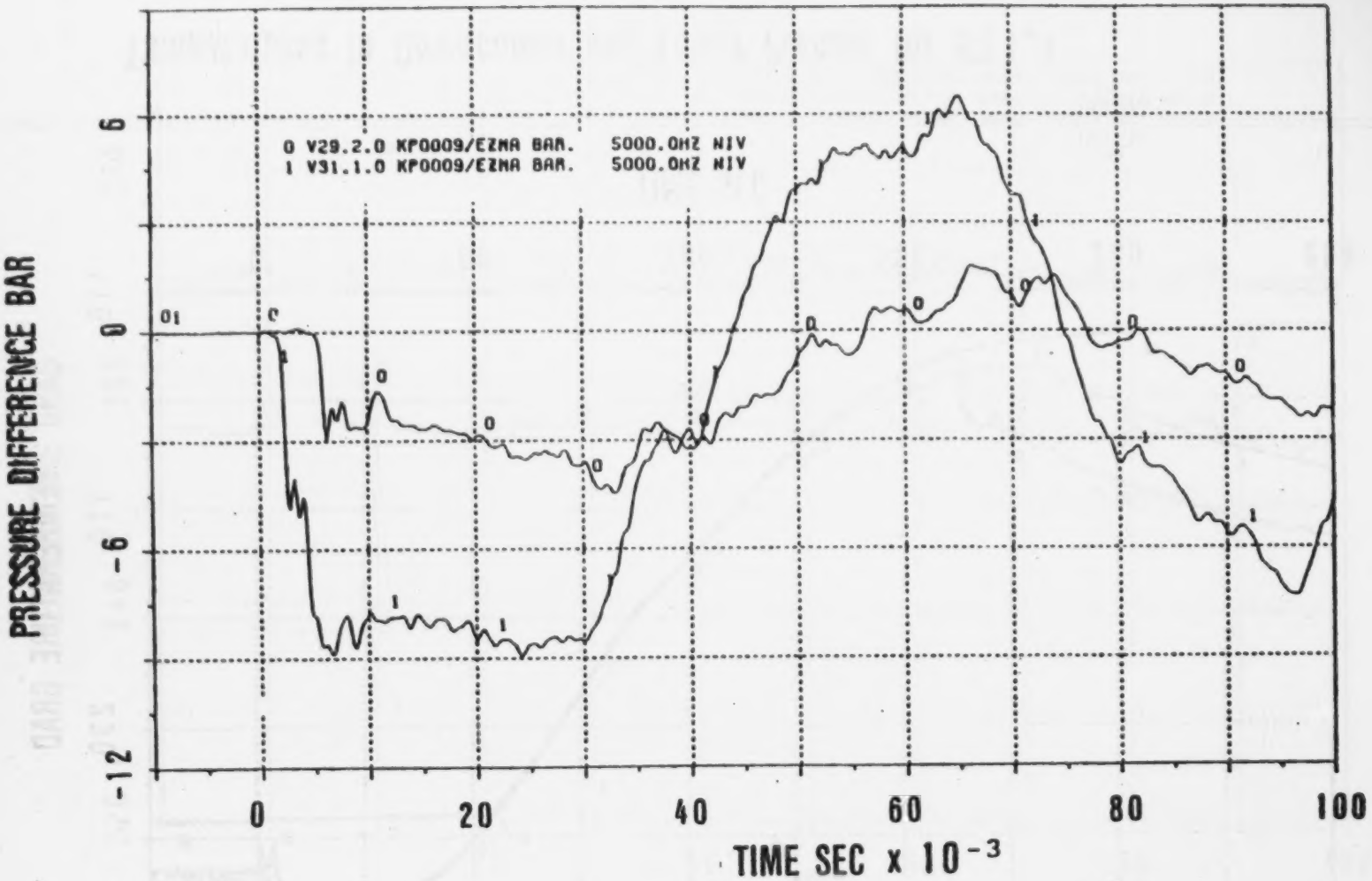
Comparison of Pressures in the Downcomes at Break Nozzle Location for V29.2 and V31.1



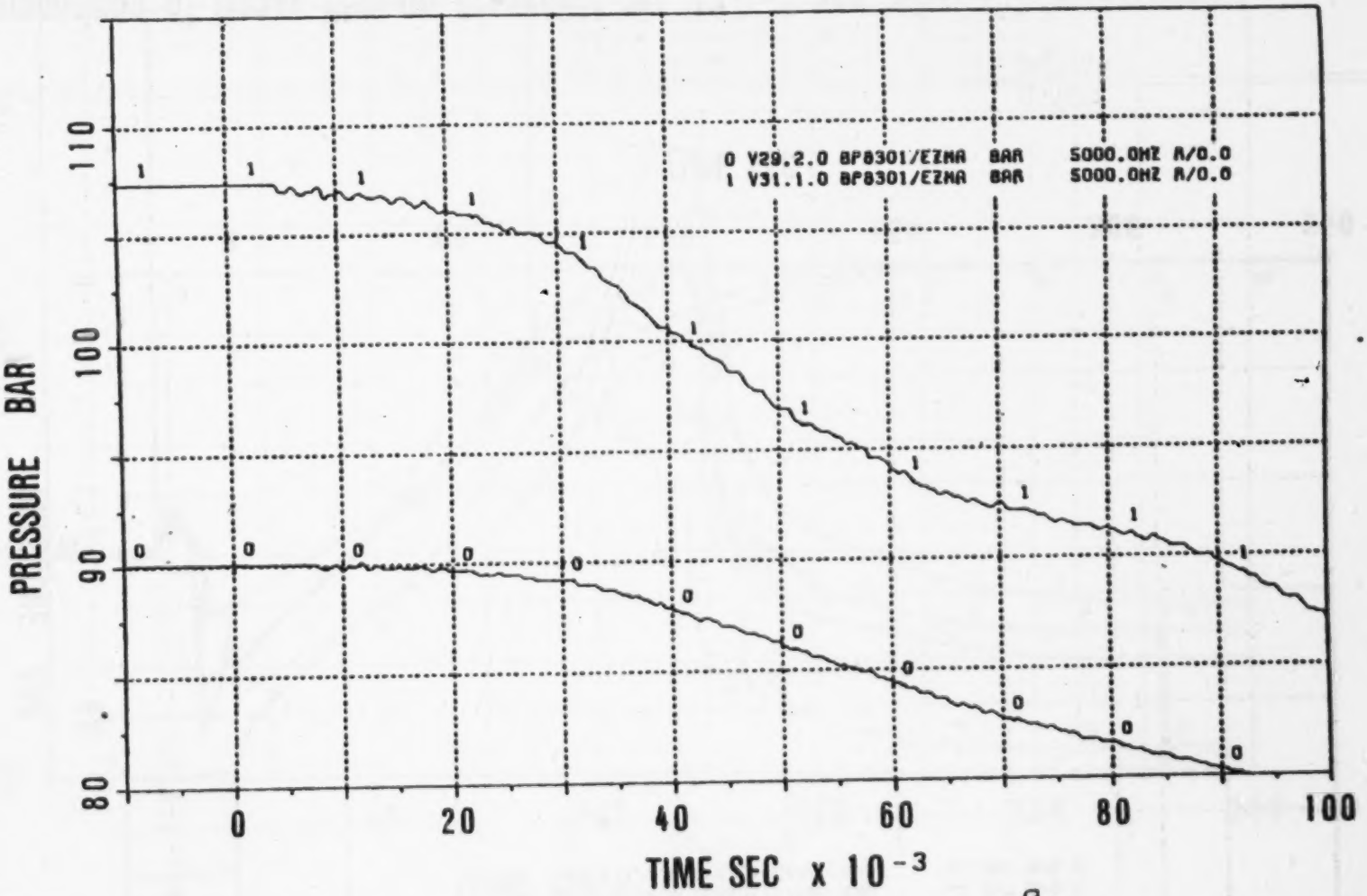
Temperatures in Downcomer and Lower Plenum for V31.1



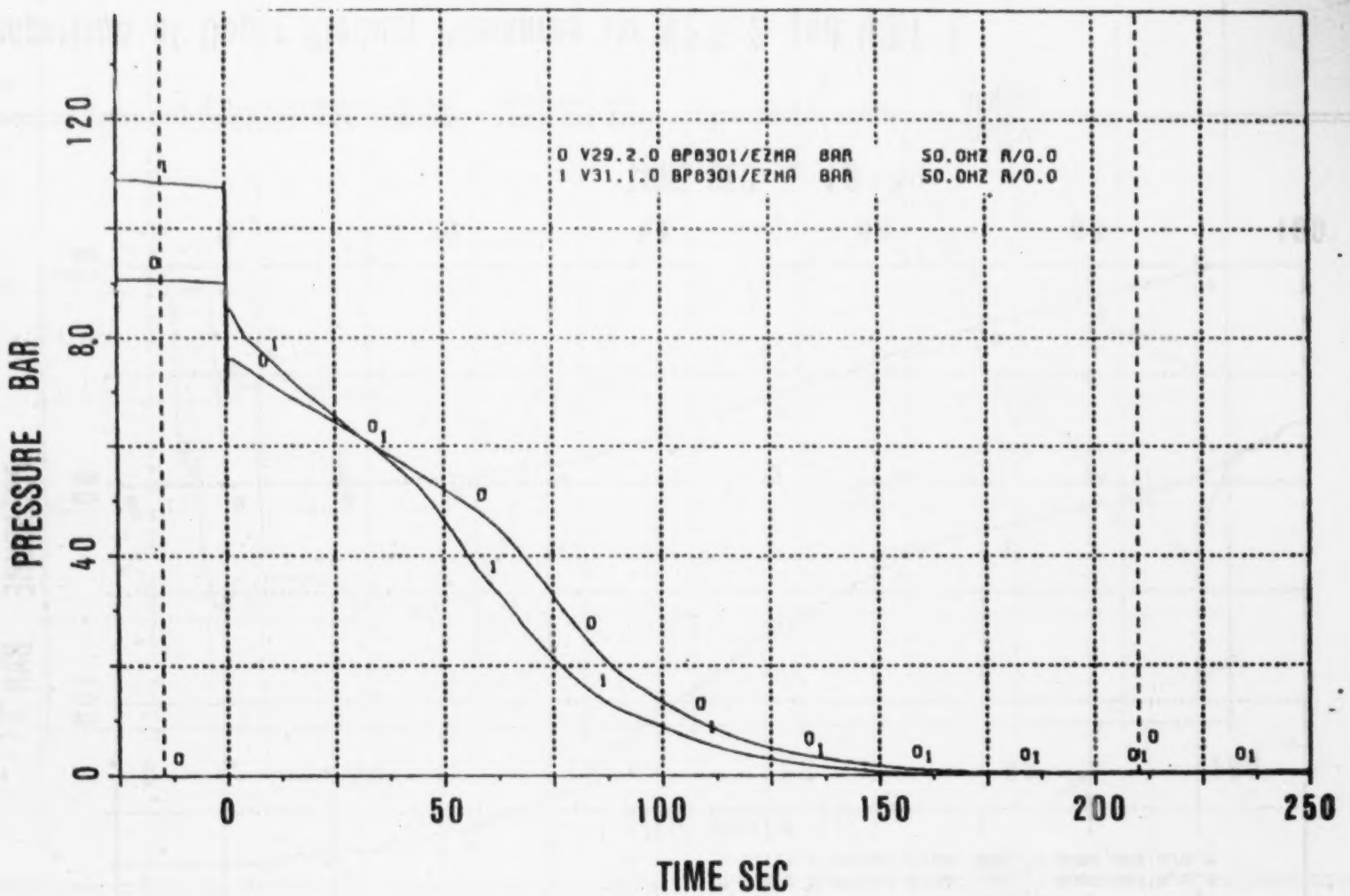
Temperatures in Downcomer and Lower Plenum for V31.1



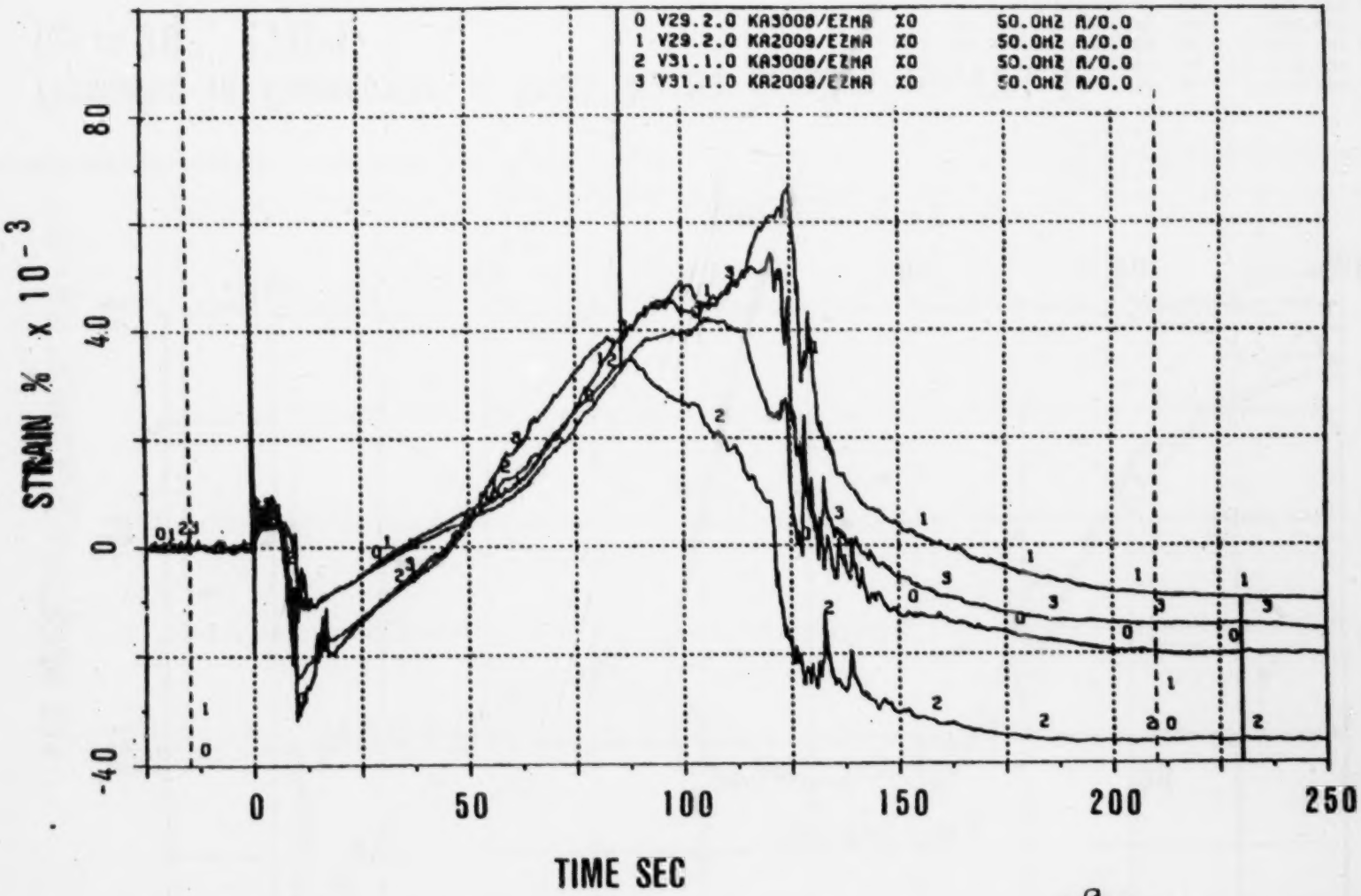
Comparison of Pressure Differences Across the Core Barrel at Break Nozzle Location for V29.2 and V31.1



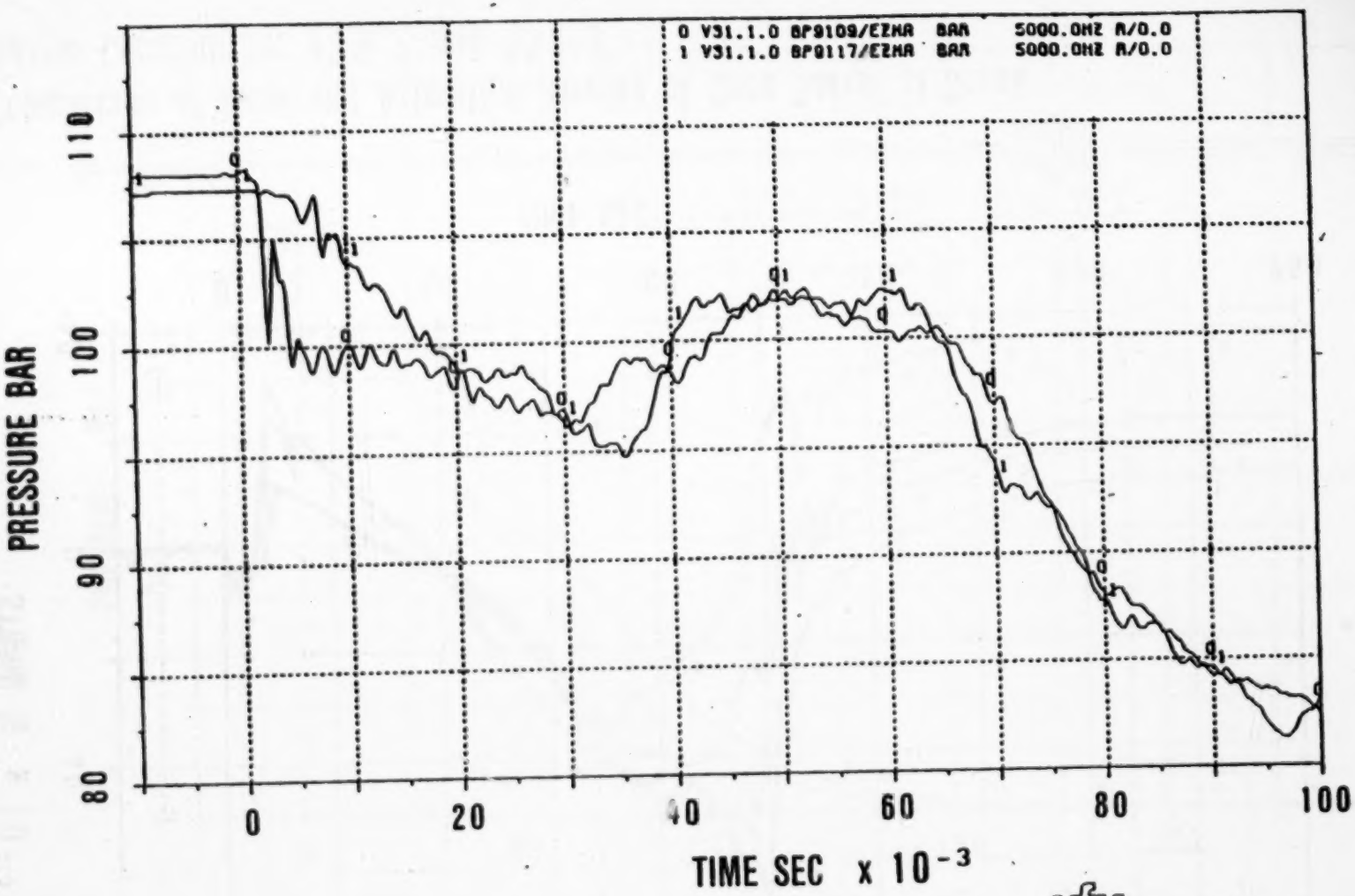
Comparison of Upper Plenum Pressures for V29.2 and V31.1



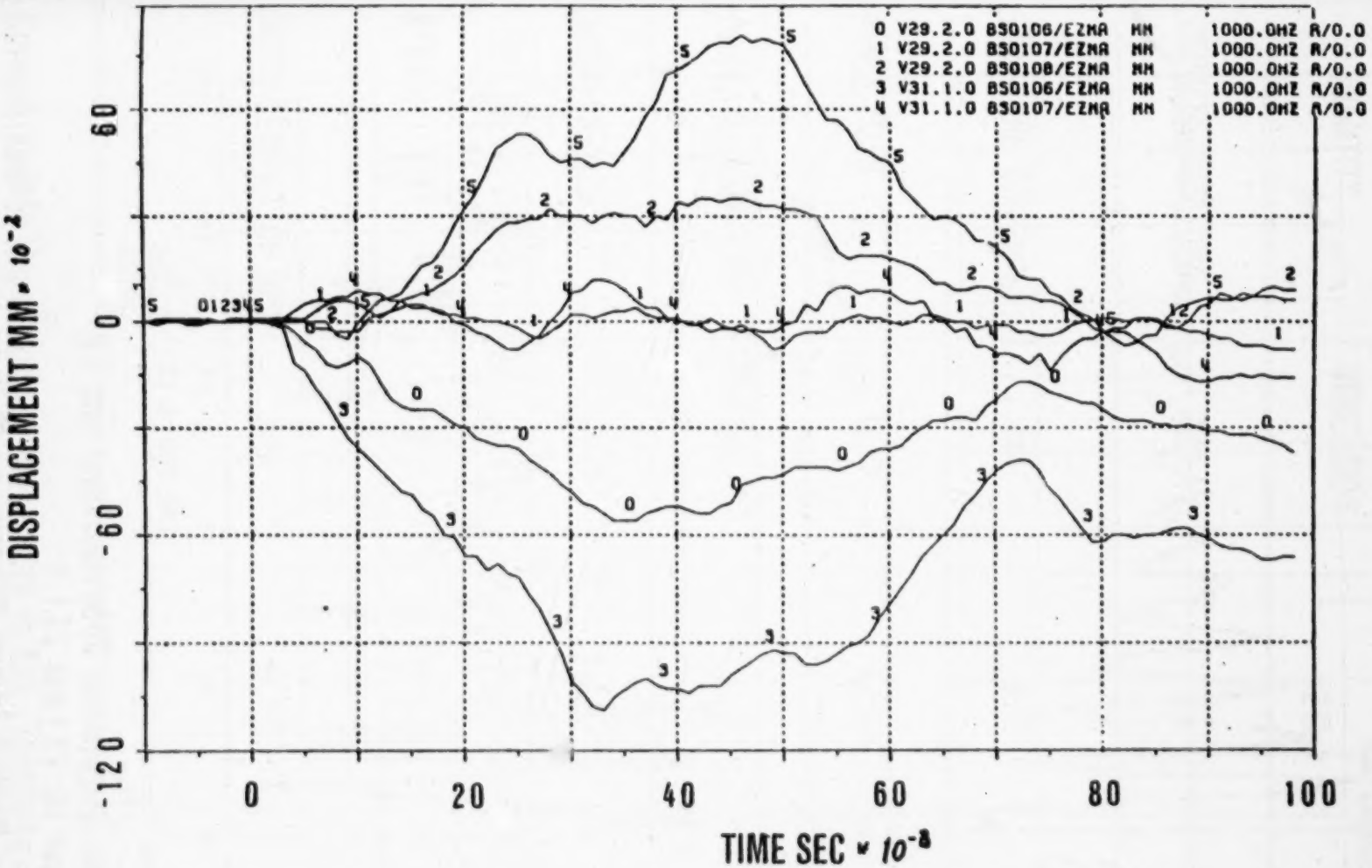
Comparison of Upper Plenum Pressures for V29.2 and V31.1



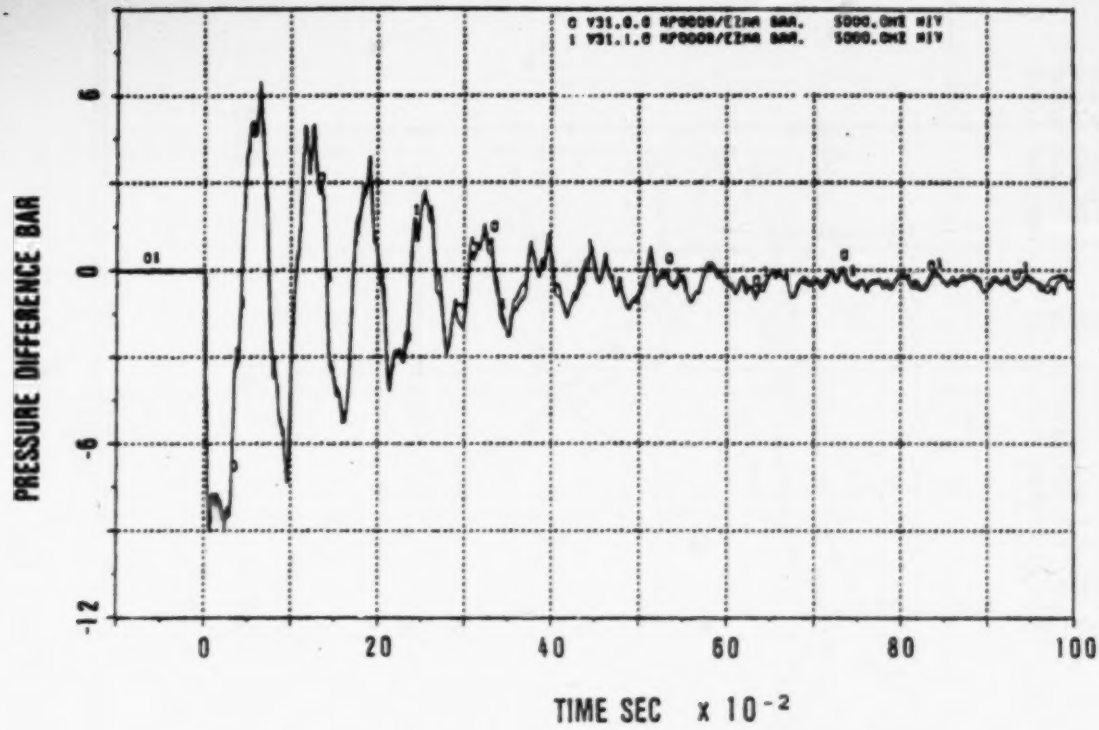
Comparison of Axial and Azimuthal Strains of Core Barrel at Break Nozzle Location for V29.2 and V31.1

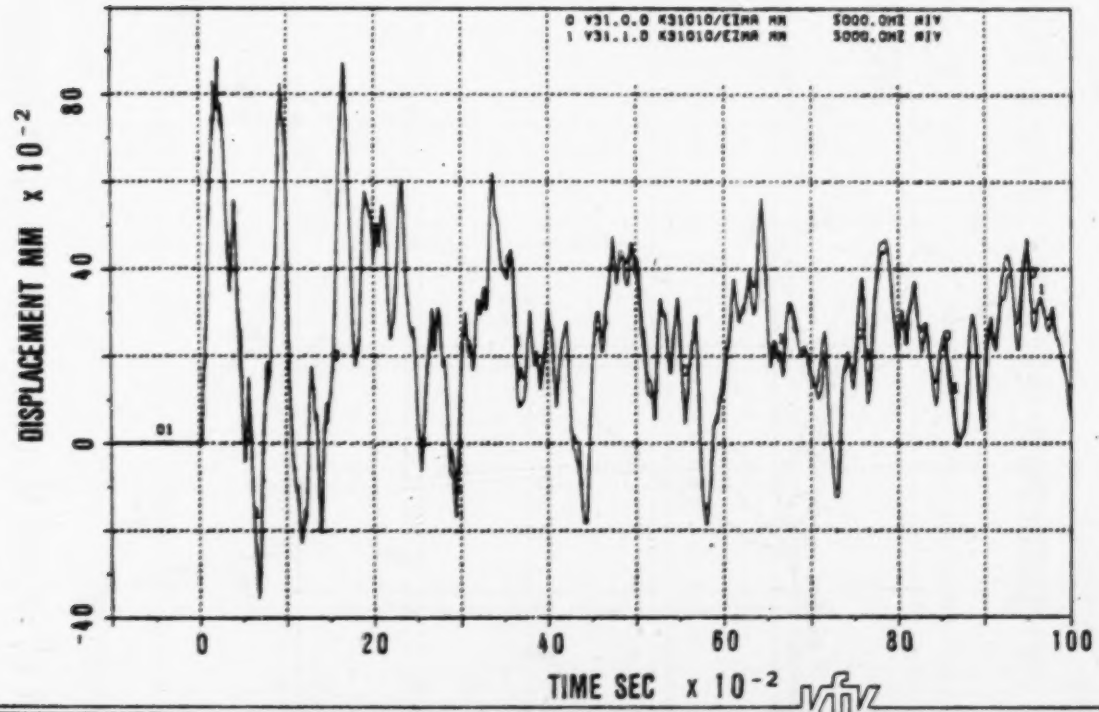


Pressures in Downcomer at Break Nozzle Location for V31.1
($\varphi = 90°, 270°$)



Comparison of Vessel Movement at Three Azimuthal Positions for V29.2 and V31.1





Vergleich von gemessenen Differenzdrücken und KM-Verdichtungen für V31 und V31.1

Anlage 5

Comparisons of Measured Pressure Differences and CB-Displacements for Replicated Tests V31 and V31.1

QUANTITY	DIMENSION	EXPERIMENT	
		V29.2	V31.1
DEPRESSURIZATION	bar	12	30
PRESSURE DIFFERENCE	bar	6.1	15.2
MASS FLOW	kg/s	1350	2320
DISPLACEMENT	mm	0.7	2.3
ACCELERATION	m/s^2	410	441
STRAIN	10^{-3}	0.15	0.4
DISPLACEMENT OF RPV	mm	0.55	1.05

KfK

Comparison of the Maximum Variations in the Fluiddynamic and structural Quantities of Tests V29.2 and V31.1 During the First 100 ms

PRECOMPUTATIONS

MULTIDIMENSIONAL MODELS, COUPLED

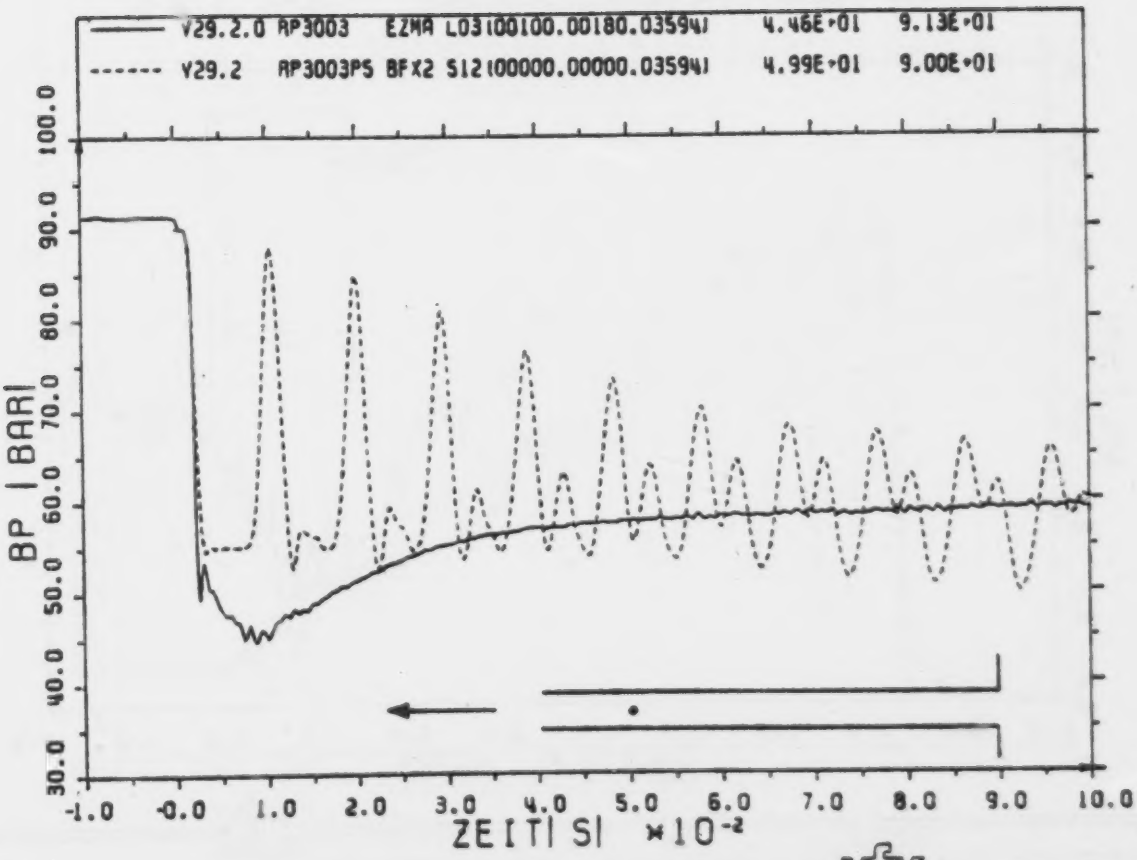
FLEXWALL,	KRAFTWERK UNION, ERLANGEN
FLUX2,	KFK (IRE), KARLSRUHE
FLUX2/DRIX-2D,	KFK (IRE), KARLSRUHE
K-FIX	LOS ALAMOS SCIENTIFIC LAB.

NETWORK-MODELS, COUPLED

DAISY,	GRS, KÖLN
MULTIFLEX,	WESTINGHOUSE

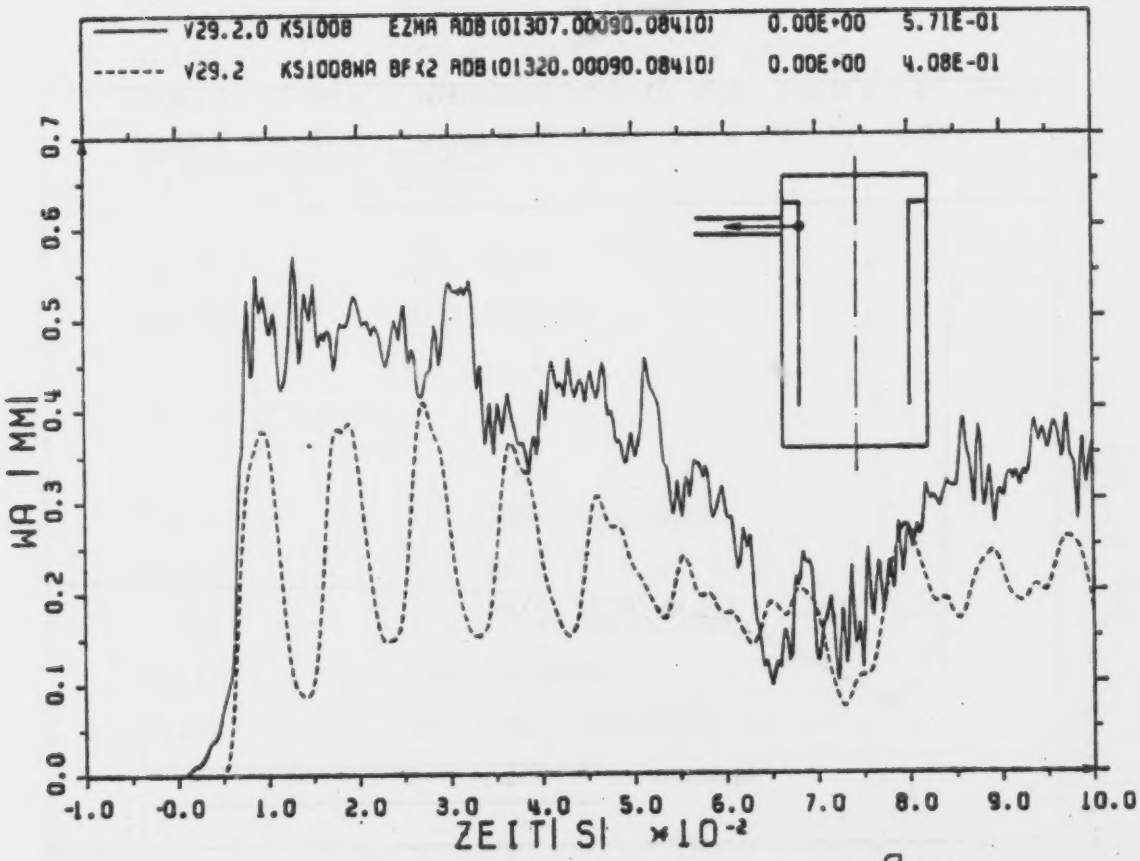
DECOUPLED MODELS

DRIX-2D,	KFK (IRE), KARLSRUHE
DAPSY,	GRS, KÖLN
FLUST,	KFK (IRE), KARLSRUHE
FLEXWALL,	KWU, ERLANGEN
FLUX2,	KFK (IRE), KARLSRUHE



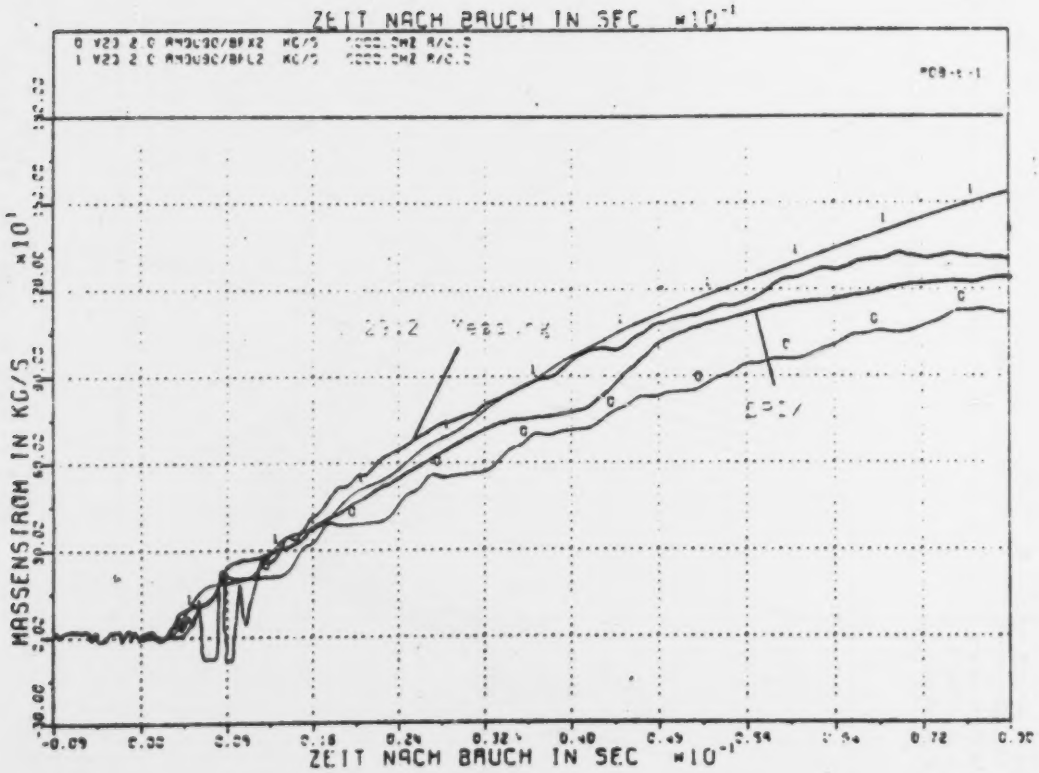
Druck im Blowdown-Stutzen,
Vergleich Messung (V29.2)-Vorausrechnung (FLUX2)
Pressure in Blowdown-Nozzle, Comparison V29.2-FLUX2

Anlage 8



Kernmantel-Verschiebung,
Vergleich Messung (V29.2)-Vorausrechnung (FLUX2)
Core Barrel Deflection, Comparison V29.2-FLUX2

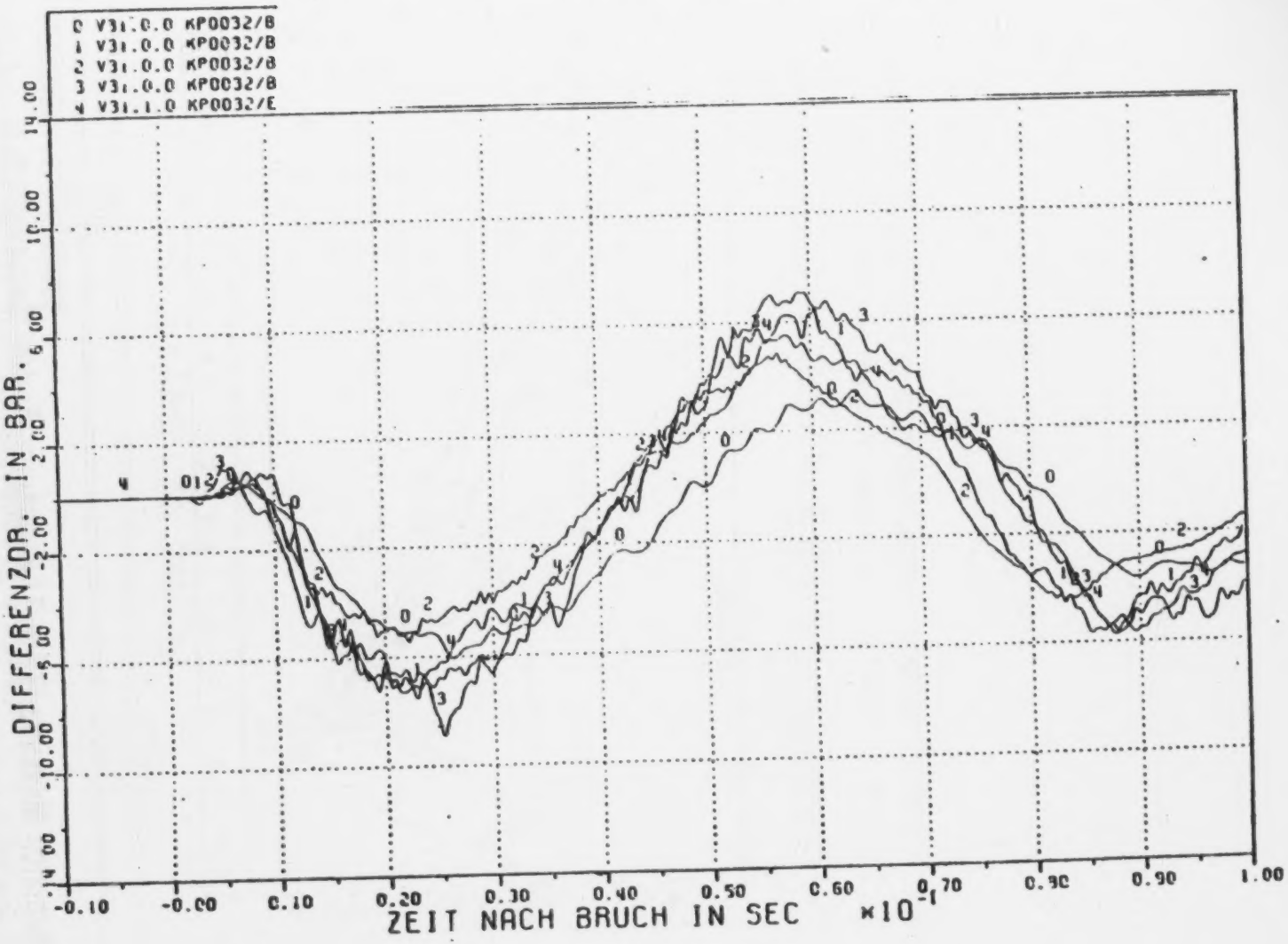
Anlage 9

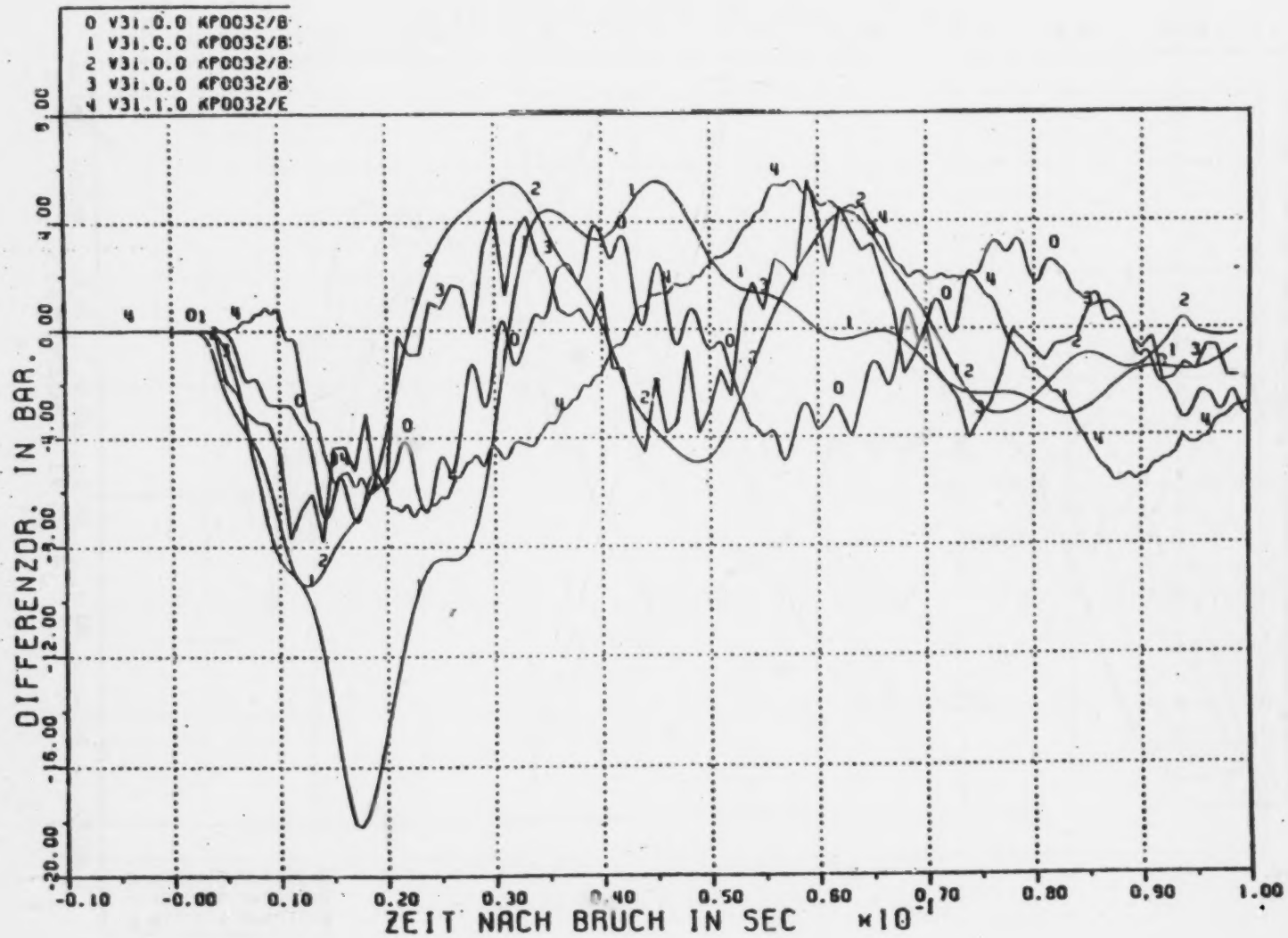


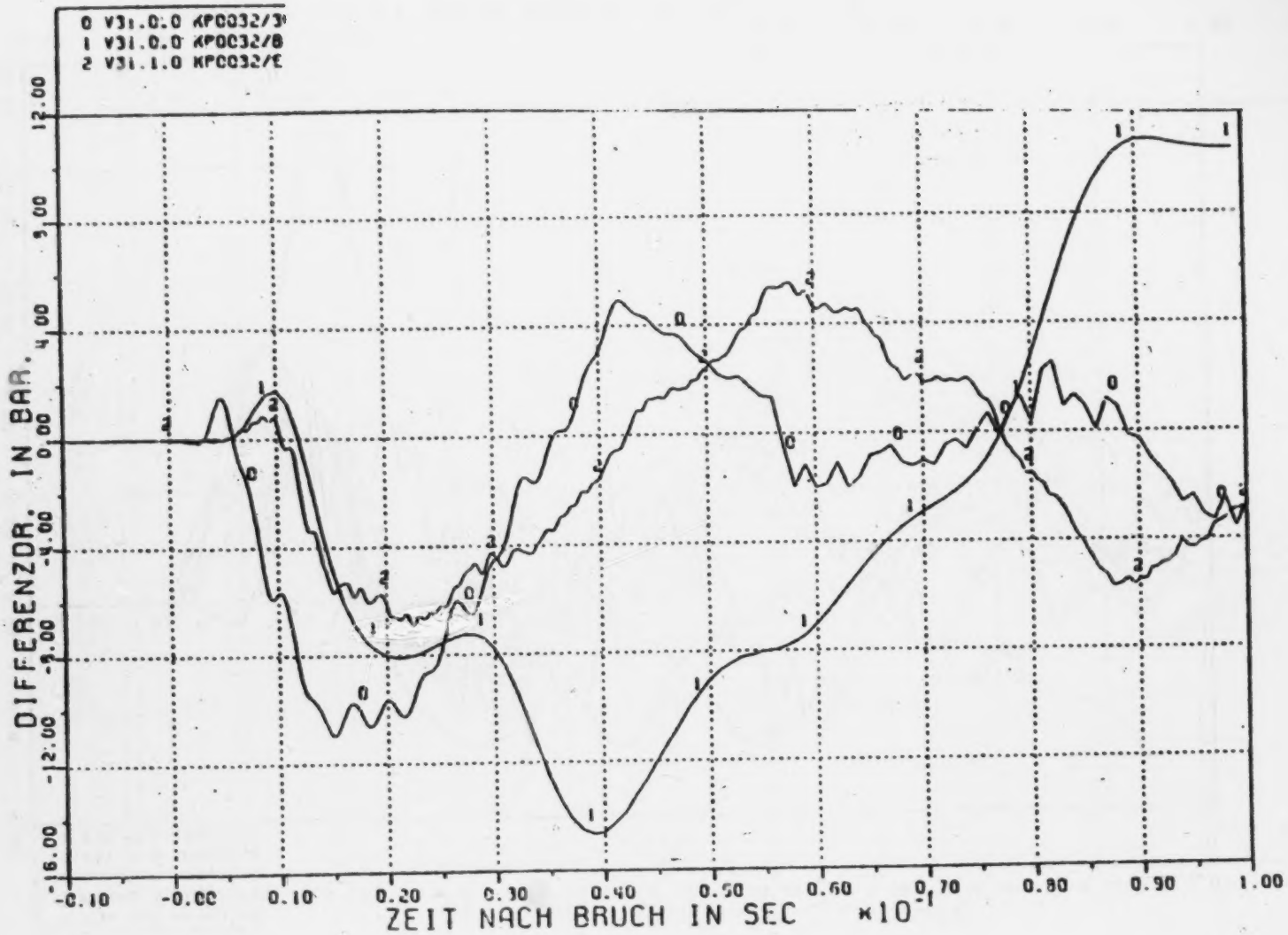
Vergleich Messung (V29.2)-Rechnung für den Massenstrom im Bruchstutzenmessring

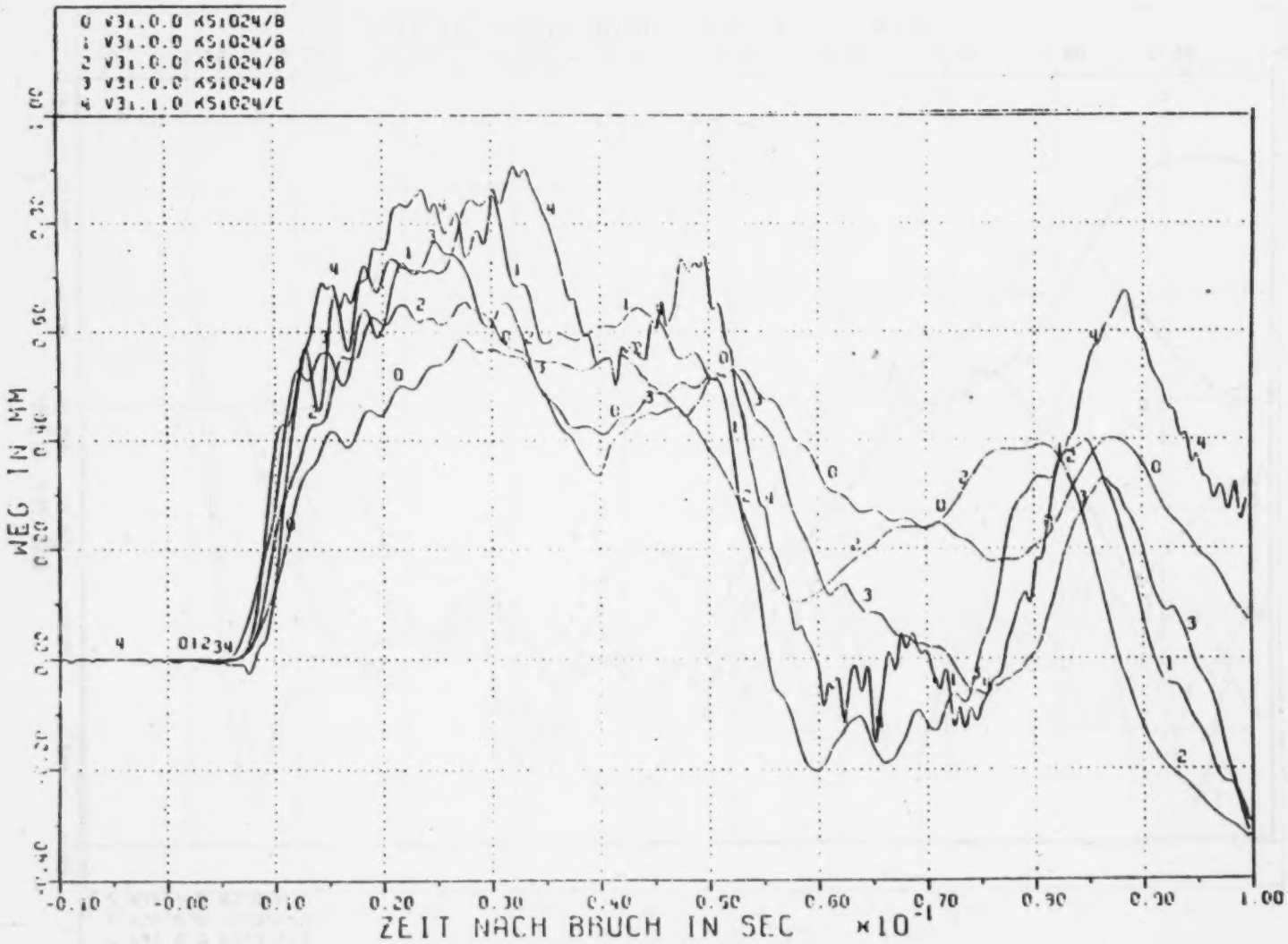
Comparison Measurement (V29.2)-Compputations for Mass Flux in the Nozzle

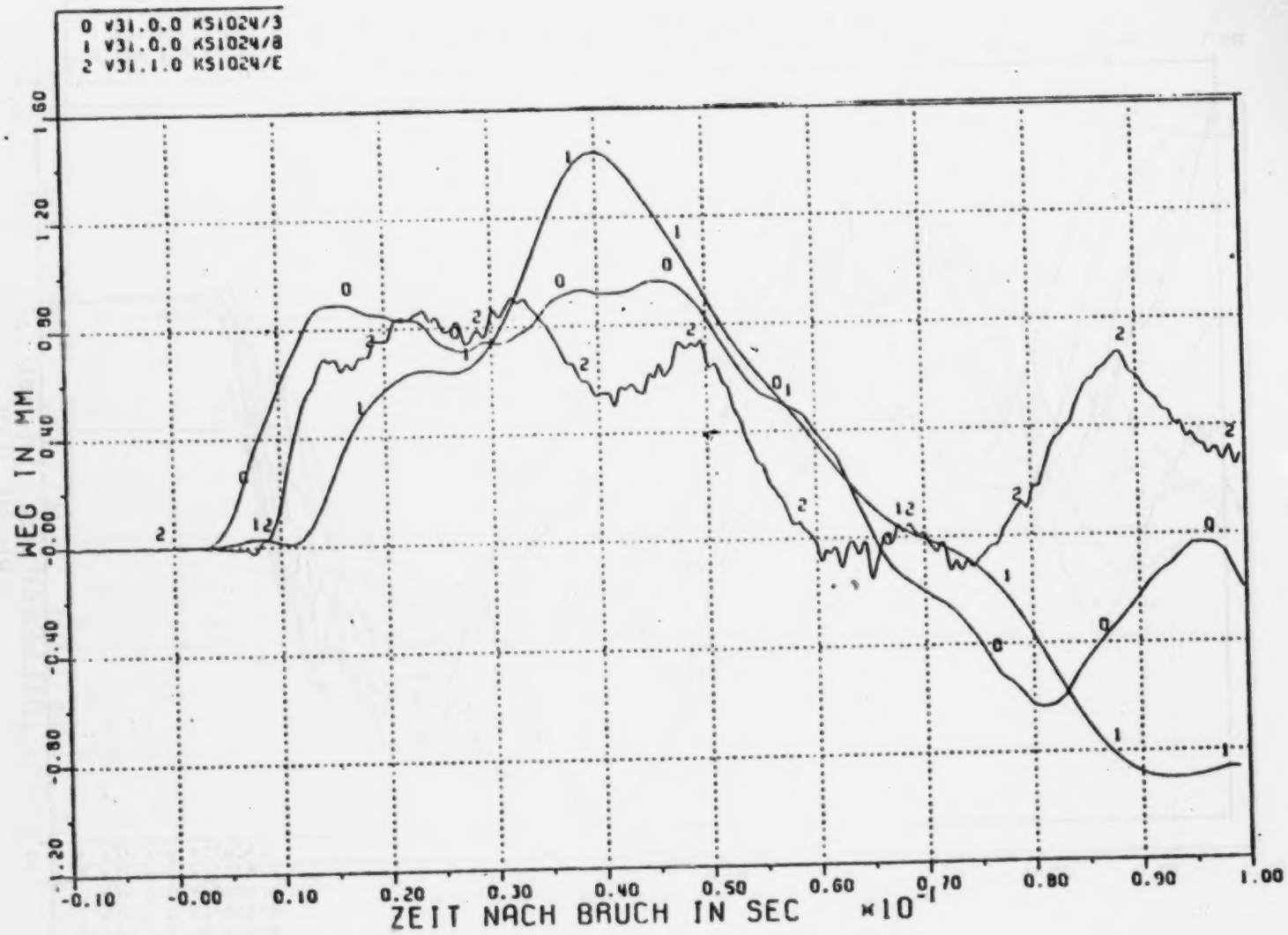
Anlage 10

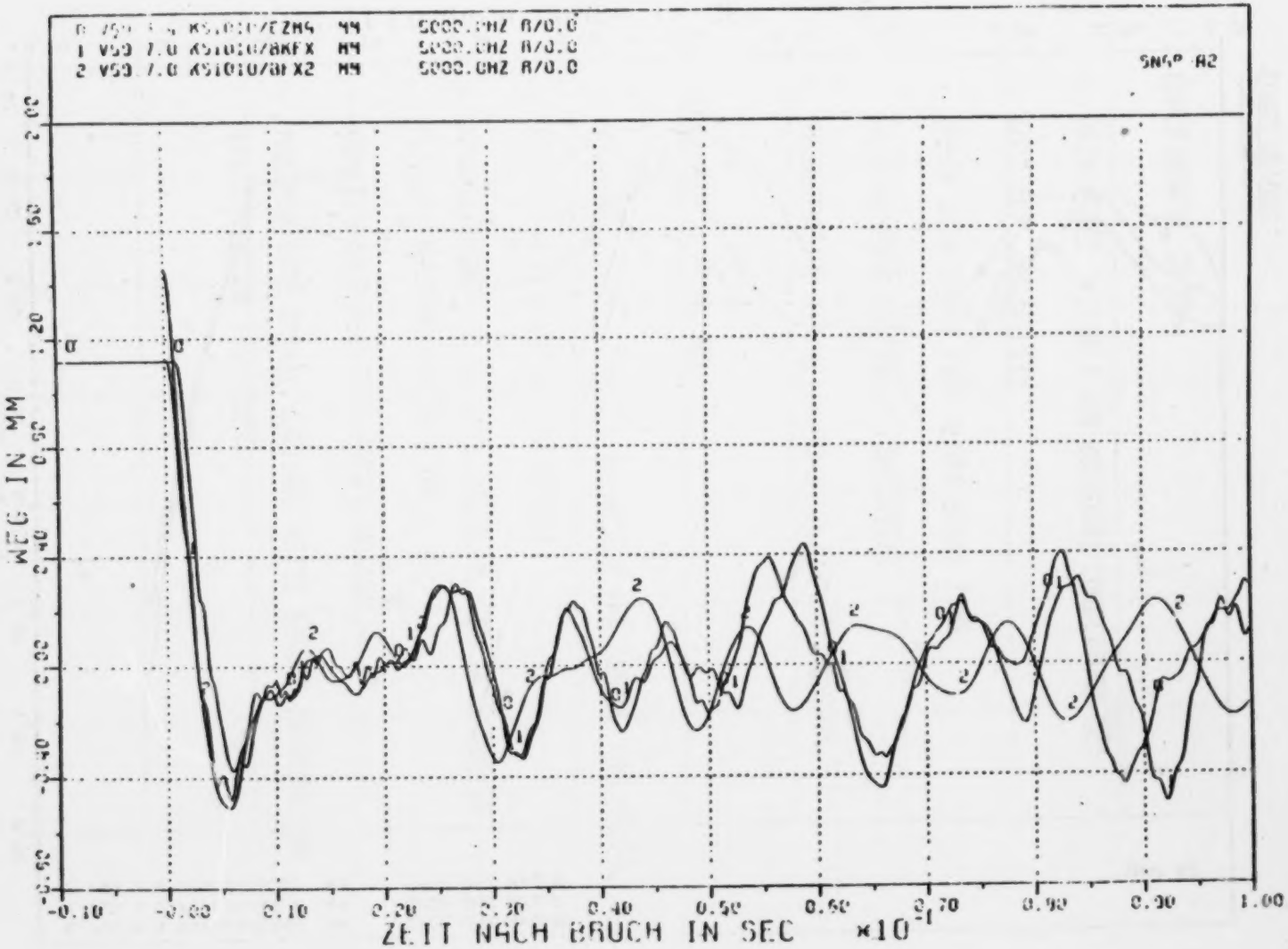

0 V31.0.0 KP0032/B
1 V31.0.0 KP0032/B
2 V31.0.0 KP0032/B
3 V31.0.0 KP0032/B
4 V31.1.0 KP0032/E
DIFFERENZDR. IN BAR.
14.00
10.00
6.00
2.00
-2.00
-6.00
-10.00
-14.00
-0.10
-0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00
ZEIT NACH BRUCH IN SEC ×10⁻¹


0 V31.0.0 KP0032/B
1 V31.0.0 KP0032/B
2 V31.0.0 KP0032/B
3 V31.0.0 KP0032/B
4 V31.1.0 KP0032/E
DIFFERENZDR. IN BAR.
8.00
4.00
0.00
-4.00
-8.00
-12.00
-16.00
-20.00
-0.10
-0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00
ZEIT NACH BRUCH IN SEC ×10⁻¹

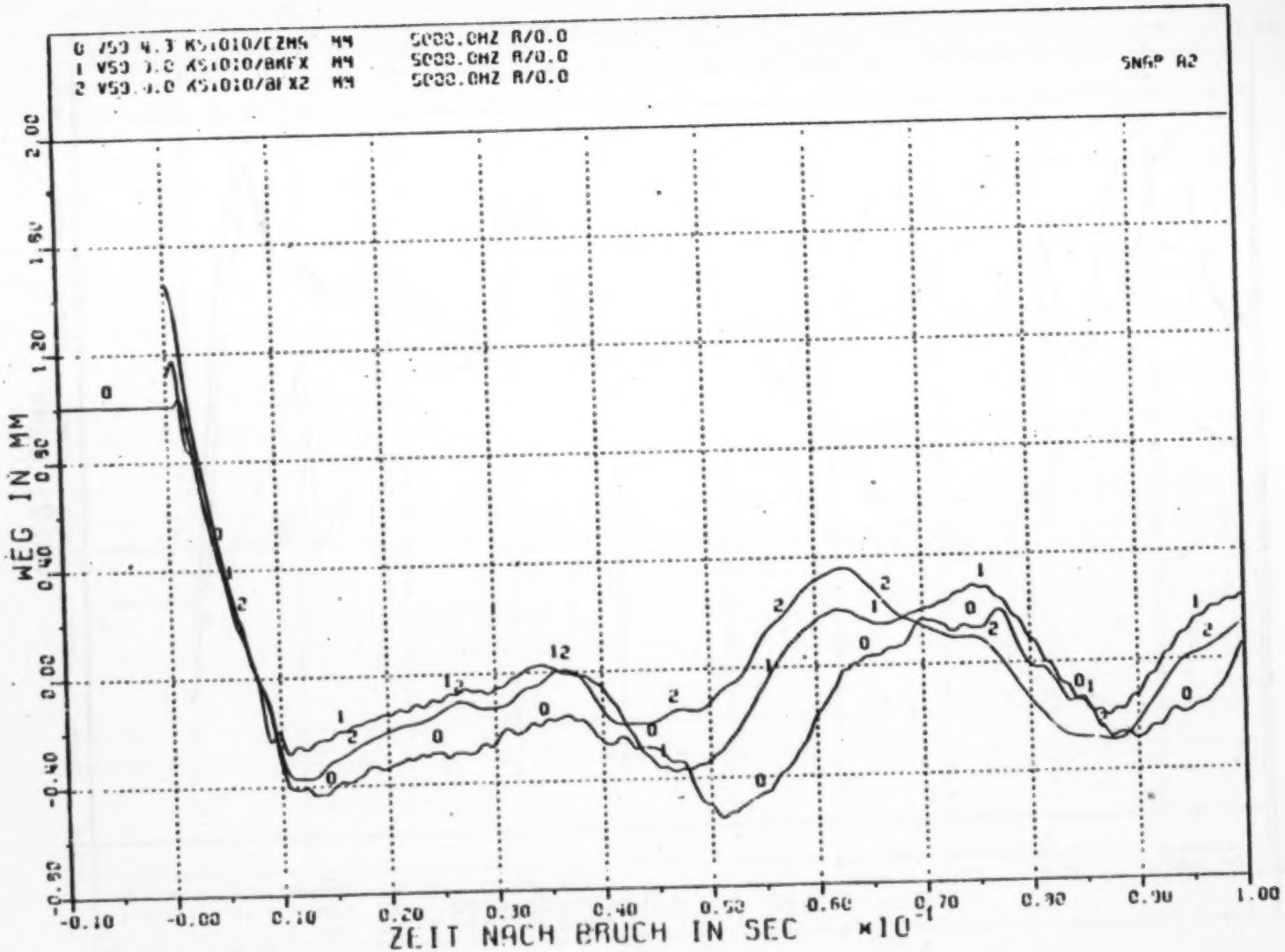

0 V31.0.0 KPC032/3
1 V31.0.0 KPC032/8
2 V31.1.0 KPC032/E
DIFFERENZDR. IN BAR.
12.00
8.00
4.00
0.00
-4.00
-8.00
-12.00
-16.00
-0.10
-0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00
ZEIT NACH BRUCH IN SEC ×10⁻¹


0 V31.0.0 KS1024/B
1 V31.0.0 KS1024/B
2 V31.0.0 KS1024/B
3 V31.0.0 KS1024/B
4 V31.1.0 KS1024/E
WEG IN MM
ZEIT NACH BRUCH IN SEC ×10⁻¹
-0.10
-0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00


0 V31.0.0 KS1024/3
1 V31.0.0 KS1024/8
2 V31.1.0 KS1024/E
WEG IN MM
1.60
1.20
0.80
0.40
-0.00
-0.40
-0.80
-1.20
-0.10
-0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00
ZEIT NACH BRUCH IN SEC ×10⁻¹



0 V59 7.0 KS1010/EZMS MM 5000.0HZ R/0.0
1 V59 7.0 KS1010/BKFX MM 5000.0HZ R/0.0
2 V59 7.0 KS1010/BFX2 MM 5000.0HZ R/0.0
SNAP A2
WEG IN MM
ZEIT NACH BRUCH IN SEC ×10⁻¹



0 V59 4.3 KS1010/EZMS MM 5000.0HZ R/0.0
1 V59 3.0 KS1010/BKFX MM 5000.0HZ R/0.0
2 V59 4.0 KS1010/BFX2 MM 5000.0HZ R/0.0
SNAP A2
WEG IN MM
ZEIT NACH BRUCH IN SEC ×10⁻¹

CONCLUSIONS

EXPERIMENTAL SETUP

- PRESCRIBED INITIAL- AND BOUNDARY CONDITIONS MET
- GOOD REPRODUCIBILITY
- BREAK DISK GIVES VERY SHORT BREAK TIME
- MEASUREMENTS GENERALLY CONSISTENT
- PROBLEMATIC: DENSITY- AND DISPLACEMENT SENSORS

EXPERIMENTAL FINDINGS

- NON-EQUILIBRIUM EFFECTS IN NOZZLE AND INSIDE THE VESSEL
- STRONG WAVE "DAMPING" IN NOZZLE
- STRONG REDUCTION OF WAVE SPEED IN DOWNCOMER DUE TO FLUID-STRUCTURE INTERACTION
- CONSIDERABLE PRESSURE VESSEL MOTION
- MAXIMUM CORE BARREL LOADINGS ARISE BEFORE EVAPORATION STARTS INSIDE THE VESSEL

CONCLUSIONS

COMPUTER MODELS

- REALISTIC ANALYSIS REQUIRES INCLUSION OF FLUID-STRUCTURE COUPLING
- MULTIDIMENSIONAL MODELS GIVE BETTER ARGEEMENT WITH EXPERIMENTS THAN NETWORK-MODELS
- TWO-PHASE FLOW IN LONG NOZZLE HAS IMPORTANT EFFECT ON CORE BARREL LOADING
- UNDERESTIMATION OF WAVE "DAMPING" IN NOZZLE CAUSES UNDERESTIMATION OF CORE BARREL LOADING
- ASSUMPTION OF
  - SINGLE-PHASE, FRICTIONLESS FLOW WITH CONSTANT SPEED OF SOUND
  - SMALL CORE BARREL DEFLECTIONS FOR LINEARIZATION OF COUPLING CONDITIONS

  ARE ACCEPTABLE INSIDE THE PRESSURE VESSEL

NEXT EXPERIMENT WILL BE A GERMAN STANDARD PROBLEM

PRESENTATION AT THE NRC MEETING 1980
WASHINGTON

TITLE: "SURVEY OF THE EARTHQUAKE SIMULATION EXPERIMENTS PERFORMED AT THE HDR FACILITY"

Authors: G. Katzenmeier, P. Jehlicka, L. Malcher, KfK/PHDR
H. Steinhilber Battelle/PHDR

Friday, October 31, 1980
Afternoon Session, Green Auditorium
Mechanical Engineering Research Programs

Table of Contents

1. Survey of HDR-Safety Program
2. Study Program Subprojekt 4000 "Earthquake"
   2.1 The Problem, Subjects
   2.2 Experimental Investigations
      - Charakteristics of Tests Performed
   2.3 Theroetical Investigations
      - Calculations
3. Central Points of Evaluation, Future Proceeding
4. Survey of Results Obtained
   4.1 Results for Reactor Building/Soil
   4.2 Results for Mechanical Equipment

Figures 1 - 15

Literature References:

/1/ HDR Safety Program Status June 1980

/2/ Technical Report PHDR 4-77, 5-78, 6-78, 10-79
"Ergebnisse von Messungen und Berechnungen der Erdbebenversuche bei niederer Anregungsstufe"

/3/ Technical Reports PHDR 13-80, 14-80, 15-80
"Erdbebenuntersuchungen am HDR, Shaker, Snapback, Spreng- und Stoßversuche bei niederer, mittlerer und hoher Anregung"
3 Quick Look Reports zu Versuchsgruppen V 63, V 64, V 65

Remark: The HDR-Program is part of an cooperation agreement between NRC and the German Gouvernement.

1. Survey of HDR-Safety Program

The investigations performed under the HDR-safety program are characterized by two features.

a) The availability of an abandoned nuclear power station allows to carry out experiments, comprising also true scale components, under actual operating and accidental load conditions.

b) The investigations are performed in a sequence in which experiments and analytics balance each other out. The aim is to process the results in such a way that they are directly applicable to the design of nuclear power stations.

Fig. 1 presents a survey of the investigations as a whole which have been performed on the HDR; the components have been set against the load cases.

The components investigated are shown in the central part of the figure and include:

- feedwater non-return and steam isolation valves, 200 mm nb, 350 mm nb, 500 mm nb
- pipework up to 500 mm nb
- reactor pressure vessel 12 x 3 m Ø
- core shroud
- safety vessel 60 x 20 m Ø
- reactor building with concrete internals and soil.

The load cases under consideration and shown in the upper part of the figure are:

- near-operation and testing conditions such as
  . conditions prevailing in a boiling water reactor
  . conditions prevailing in a pressurized water reactor
  . hydraulic tests

- loads resulting from an accident such as
  . loads due to blowdown with free jet
  . thermal shock loads
  . earthquake loads
  . loads due to aircraft impacts.

The purpose of investigations is, on the one hand,

- to record and understand the fluid thermo- and structural dynamic loads as well as the resulting reaction and component loading,
  and, on the other hand, above all

- to verify design methods presently applied for the components and load cases indicated.

The following contribution will deal with the experiments and calculations made on the earthquake load case.

2. Study Program SP 4000 "Earthquake"

2.1 The Problem, Procedure, Subjects

In the computational analysis of the vibratory behavior of nuclear power stations exposed to seismic or other impacts mostly linear methods are presently used which are capable of reflecting reality by a limited approximation only. It is known and has been demonstrated in the HDR-tests that effects are generated, mainly by large excitation forces, which cause non-linear behavior of vibration. Such effects might be, e.g.:

- Modification of boundary conditions such as clearance, deformation, etc.
- Change of material characteristics such as plastification of steel, behavior of the soil, change of attenuation.
- Change of mass portions and stiffnesses.

The increase in damping associated with a growing ~~number~~ of deflections, on the one hand, constitutes a desirable conservative characteristic of linear calculations but, on the other hand, the non-linearities might give rise to shiftings in resonances and phases, which are susceptible of adversely influencing the defined harmonization of structures.

It is necessary under these aspects - and we consider this item as the principal targets of HDR-investigations - to deepen our knowledge of the vibratory behavior of real structures and to validate it further in order to find out above all which computational methods and which boundary conditions allow to record the actual vibratory behavior at high excitation and with sufficient accuracy.(It should be noted here that although preliminary considerations have been made regarding the limits implied by the term "sufficient" in connection with risk analyses, no objective, quantitative criteria have been available so far in practical application so that judgements have to be made mainly from the engineer's point of view.)

The following blocks for investigation result from this problem:

a) Experimental analyses of the vibratory behavior and the loading of the HDR and its internals at earthquake type excitations.
b) Theoretical analyses based on different methods of computation.
c) Comparison and judgement of the computational methods and of boundary and input data.

The following investigations proved to be necessary in addition:

d) Experimental "in advance" investigations for determination of the realistic material parameters and boundary conditions, above all characteristics of the soil, pipework hangers, material data of the structure of the building, etc.

2.2 Experimental investigations

The concept as well as the procedure in experimental investigations, in particular

- the excitation modes and their attribution to
- the components under investigation

will be elucidated by the following figures. To begin with, I wish present in brief by Figs. 2 and 3 some data concerning the real size and geometry of the HDR and the components under investigation. The containment shown in Fig. 2 is a highly compartmentalized, fully pressurized containment. Its height is 60 m, its diameter about 20 m. As appears from Fig. 3, the pressure vessel is 12 x 3 m in size, the wall thickness is 100 to 140 mm. The nominal bore of the pipework is 450 mm inner diameter at the maximum. The HDR-Plant is built by german rules.

Excitation methods:
The reale earthquake excitation takes a time of about 10 - 30 sec with acceleration values of maximaly 1 to 2 m/s² (german rules) in the frequency-area of 2 - 5 Hz. Anexact realisation of these conditions is not possible mainly because the necessary high excitation energie. Therefore the excitation was performed by the following methods:
shaker, snapback, explosion and by solid propellant rockets in addition (see picture 4)

The tamped explosion gives the best approximation of a real earthquake load-characteristic relative to the height and the frequencies of the acceleration, but not in relation to the duration. Fig. 5 makes clear that the load spectrum generated in the experiment - value of excitation as a function of frequency - agrees quite well with the soil spectrum as specified in the U.S.A. by NRC for the design of nuclear power stations.

The Shaker (back to Fig. 4) produces with two excentric rotation masses a sinusoidal load. By varying the frequency and size of loading both the intrinsic behavior and the imposed, load-depending vibrations of the bulding and the mechanical structures can be excited. Especially it is possible, to excit all important resonance-points of the structures, therefore we could reach high-excitation levels.

Smaller shakers (b) used fulfilled the same tasks, i.e., direct excitation of the pipework and tanks.

The snapback tests (Fig. 4, bottom) - with a quasi-vertical load characteristic -, as a supplement to the shaker tests, are suitable for high and pronounced excitation of the low resonances of pipework. The height and the direction of the excitation is variable.

In addition to these tests, preliminary tests were made with a load by excitation on the reactor building generated by solid propellant rockets. Fig 5 shows the nearly rectangular course of load versus time (400 KN with 1/2 s duration). This test configuration allows to investigate the global reaction of the building and the excitation of

mechanical equipment of the reactor under a load similar to that in an aircraft impact. The rockets are fixed at the outer side of the containment.

Survey performed tests and measurements (Picture 6)

In picture 6 the complete investigated structure-chain is shown with the gleets:
- soil, building, RPV, RPV-Internals and pipes.
Involved in the system are all interaction-tasks. Further measurements are made for another building with a thin-wall, high, waterfilled flood water storage tank (FWT) and also for the neighbor-power plant VAK. These components are not drawn in picture 6.

In the explosion tests the dynamic behavior of the whole plant from the soil up to the mechanical components was excited and investigated. Especially the soil/structure and also the building/components interaction can be studied. The excitation-level - qualitatively marked by the vertical column on the horizontal arrow's - was relatively low for every components. It was used 5 kg explosive as maximum, because of safety requirements for the Kahl Nuclear Power Station nearby which is still in operation. Picture 6 makes clear, that the explosion can simulate the real earthquake-excitation approximatly for all the system-parts, except the hight and the duration of the excitation-force. Therefore we use the other excitation methods for a higher, local excitation of the plant.

The shaker tests comprising excitation of the building were performed at medium level of excitation. The maximum response measured attained 0.64 m/s^2; the response of the indirectly excited pipework amounted to approcimately five times this value. In 6 of 19 tests the system was heated to conditions prevailing in a boiling water reactor (70 bar, 285°C).

The tests involving the smaller-scale shaker directly installed on the components were performed at clearly higher excitation. They gave rise to relatively large movements of the pumps and pipework. The values of acceleration attained about 30 m/s^2. The shaker tests allow first of all the investigation of the relation between the height of excitaton and the vibration behavior and therefore to find the non-linearities.

The snapback tests reached, as shown in picture 6 - the highest local acceleration. These excitation was used to find first of all the low values of "Eigenfrequencies" and the damping values of the components.

The rocket test excited building, RPV and pipes, the highest response in both test was 0.6 m/s^2 in the pipework. These experiments was performed first of all to test the excitation method on a real plant and also to study the global response of the building and the response of the reactor internals. To record by measurement the vibratory behavior and the stresses on the building and mechanical equipment a lot off acceleration transducers, displacement transducers, powered picups and strain gauges were installed. (Fig. 8). For evaluation,

i.e., for the determination of the natural frequencies, natural modes, damping values and also of the vibratory behavior imposed, data on acceleration are mainly used and the time histories allow to derive the frequency spectra which are used for evaluation (some more information to the test in picture 7).

2.4 Theoretical investigations

As already mentioned, the measurements primarily serve the purpose of verifying the computational methods; for this reason, extensive calculations are being made in parallel with the experiments. We make a distinction between the following types of calculation (Fig. 9)

- design and safety calculations for optimizing the experiments and the safety of the plant;
- pretest-calculations of the natural frequencies, the mode shapes and the dynamic response of all the components for the comparison calculation/measurement (verification
- post-calculations, dependent on the comparison of measurement and calculation for indentifying individual parameters.

Fig. 8 offers a survey of the calculations made for the whole structure-chain and all load conditions. The listing relates to the last group of tests performed at medium and high excitations. The precalculations were made by two independent groups, one industrial group and a so-called "regularity group". Not yet included in the list is the group of international partners. A number of linear and non-linear computations have been performed,

with different modelings. The codes used bear familiar names, e.g., ANSYS, SAP, STARDYNE, etc. (more details in Fig. 10).

The calculations are made in the real practical way step by step with all problems of coupling and load-propagation. Therefore an important issue in our verification-process is to control and to study the methods and the calculation-data in all coupling/transition points to detect errors and first of all to find out the influence of different parameters and modelling-techniques.

3. Main Points of Evaluation, Future Proceeding

The previous experiments performed at low, medium and high excitation have in our opinion furnished the proof that the models of excitation apllied, i.e.,

- shaker at the reactor building, reactor pressure vessel and pipework,
- snapback tests on components and pipework,
- explosive excitation with charges tamped in the soil,

are able to simulate earthquake-excited loading conditions and the levels of excitation chosen are suitabel to record the vibratory behavoir of the building and the structures. The desired variables-natural modes of vibration, natural frequencies, damping values, vibration histories at external excitation - have been determined and are available together with the data from the pre-calculations mentioned before.

The comprehensive evaluation including post-calculations concentrates on the items (Fig. 11a):

- determination of the paramters of the real structure
- description of the real structure by a model
- possible better procedures for the description of the real vibration behavior.

The main tasks in these areas - shown in Fig. 11 - are the following we give highest priority in our work:

(1) Recording and numerical description of the damping behavior of all elements of the system.

(2) Suitable modeling at reasonable costs, i.e., idealization (e.g., beam, shell model, continua and discretization).

This is closely connected with the questions:

(2a) In which cases is a coupled or decoupled calculation correct or admissible?

(2b) Regarding above all the soil/building complex: In which way must the propagation of load be taken into account?

Other points of primary interest are:

(3) The influence and analytical treatment of the boundary conditions and supports, with the question of stiffness (3a) for mechanical equipment and soil/building interaction (3b) for the building standing in the foreground.

(4) The actual stresses,especially on pipework and vessels, will be recorded in detail and investigated in a comparison of measurements and calculations.

(5) In addition to the individual problems the question must be asked above all which computational method (linear, non-linear, quasi-linear) will allow to analyze most favorably and reasonably in terms of economy a case under consideration, in this particular context the pipework. This is linked with the attempt,

(5a) to develop and use appropriate and adequate methods of approximation instead of expensive, hardly transparent FE-models. However, the range of validity must be strictly delimited against the experiment.

To deepen the quantitative evaluation of individual items, e.g.,

- influence of elbow modeling, damping with and without inner pressure additional tests in a test bench will be performed under the HDR safety program on a part of the pipework system taken from the HDR; see Fig. 12. The vibratory excitation is generated by shaker and snapback as in the HDR and, in addition, by hydraulic excitation with an earthquake type load history.

4. First Results Obtained

Having described the experimental, measuring and computational techniques and the tests performed with the intention to present at the same time a rough survey of data available so far, I wish to give now some results by way of example without going into detail.

4.1 Results for Reactor Building/Soil

As regards the building, about 15 natural frequencies and modes of rocking, translation and torsion have been identified already in the tests performed at low excitation in the frequency range up to 30 Hz. Fig. 13 shows the most important natural modes measured for the building (modes of rocking). Calculations relating to the building and based on about ten different FE-models, both

- beam models (40 - 300 degress of freedom) and
- shell models ($\doteq$ 10.000 degress of freedom),

and different discretizations were able to reflect reality partly well and partly with restrictions only.

The following statements were derived from the comparison of measurements and calculations:

- As regards soil modeling, a consideration relying on real coupling of soil and building does not yield visible advantages. Replacement of the soil by soil springs according to the half-space theory, attaching the soil/building at the interface, seems suitable and adequate.

- Regarding the models of the building it became clear that the vibratory behavior of the building can be described with the same result or even better by a simple but good beam model, compared with the more expensive fine-mesh FE-models based on plate, shell or volume elements. This statement is illustrated by Fig. 14.

- As regards the computational method it became evident that direct integration (THA), modal analysis (TMMA) and the fast Fourier transformation (FFT) are equal in right in the linear range.

- At high excitation we find distinct non-linear effects, e.g., the first bending mode moved downwards with increasing level of excitation. This is probably an effect of the non-linear soil-shear modulus.

3.2 Results for Mechanical Equipment

For the reactor pressure vessel and the recirculation loop noticeable deviations of measurements and calculations occurred in the shaker tests, which are attributable to considerable obstructions to vibration by hanger elements. For the tests involving high snapback and shaker excitation the fundamental mode behavior of the components (primary steam line, recirculation loop, flooding water tank) has been so recorded to give largely accurate values.

The calculations, especially those on low-level excitation, have not yet produced a uniform statement concerning the efficiency of the different models with respect to the components, especially with respect to the question of coupling/decoupling. Generally, the following statements can be made, given the present status of evaluation:

The quality of a model for the components installed in large systems is not determined primarily by the number of degrees of freedom.

Appropriate, simpler modeling derived by engineering, with decoupled systems, is capable in many cases of furnishing sufficiently good results and, above all, transparent data.

As an example of a result obtained from the second test series performed at high excitation I wish to mention a comparison between cold and warm conditions. Two shaker tests were carried out, as shown in Fig. 15, under similar load conditions, excitation of the building, indirect excitation of the pipework (primary steam line).

The pipework connections (508) do not exhibit differences, which is according to expectation (good reproducibility). By contrast, we have a clear difference as to the responses in the center of the pipe. Under hot conditions a dominating peak was observed, whilst in the cold condition two smaller peaks occurred with the same mode. The cause lies probably in a restriction on pipe deflection.

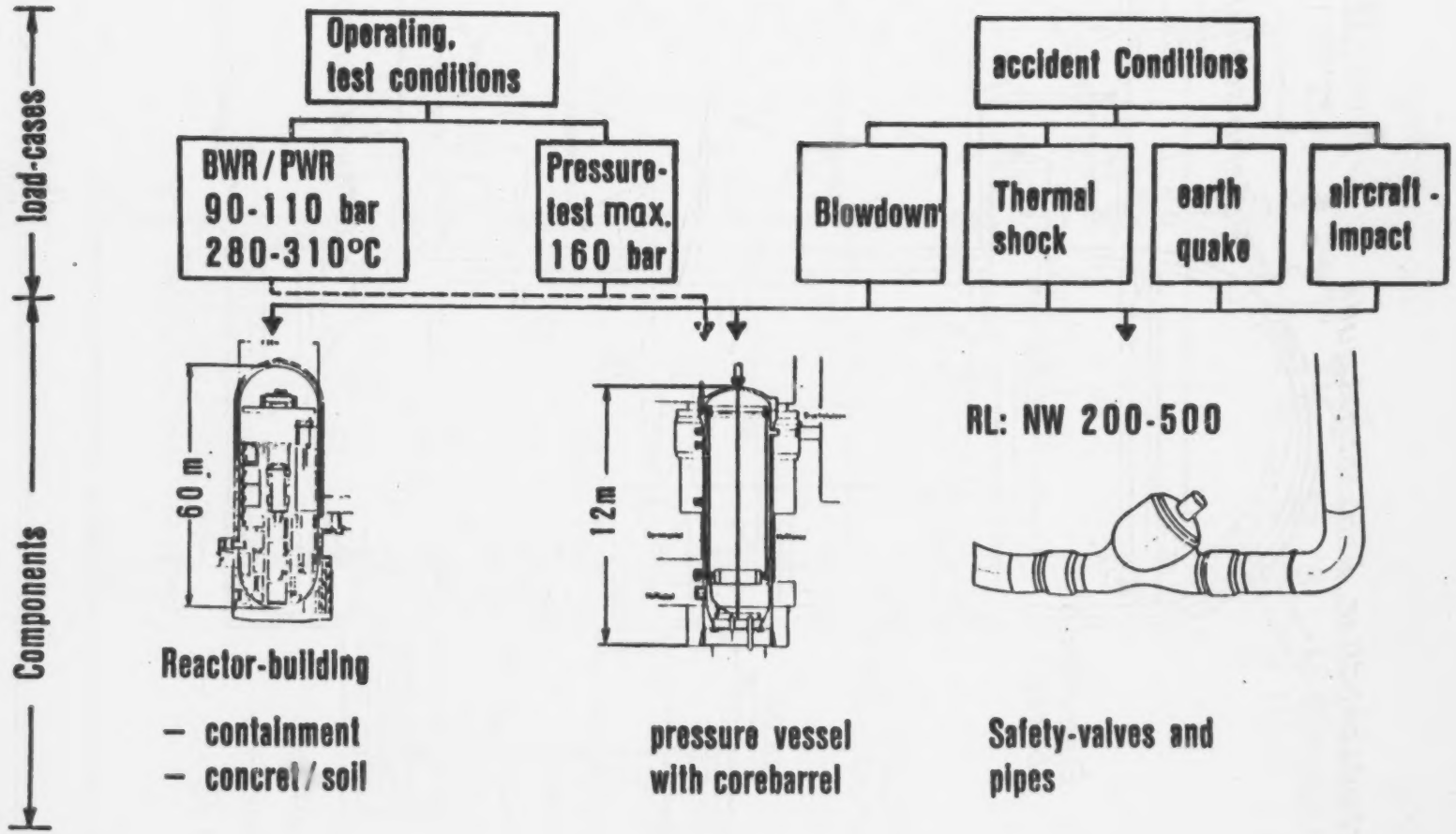
The evaluation of the HDR tests as beginning "a part" is under way; we expect in 1981 significant, quantitative statements regarding the questions indicated.

Like all HDR-experiments, the investigations will be performed in close cooperation with several institutions. As already said, the following institutions have cooperated in the seismic studies:

- on the German side a number of independent institutions and firms;
- on the American side at present mainly ANCO, INEL and ILL.

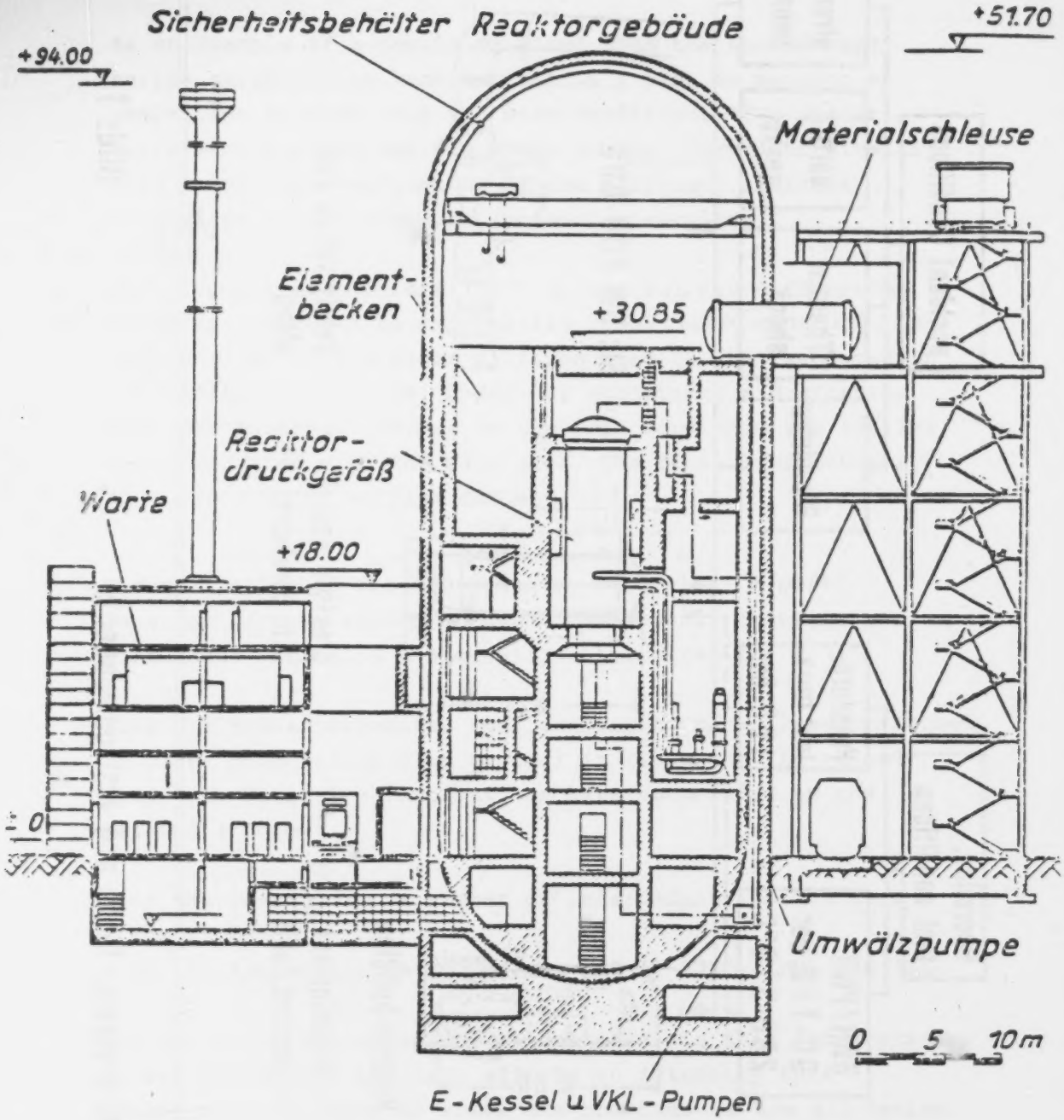
The following contributions will illustrate this cooperation. We are interested and take efforts to intensify this cooperation in order to solve the problems we are all facing.

Thank you for your attention.



HDR-program: loads and test-components

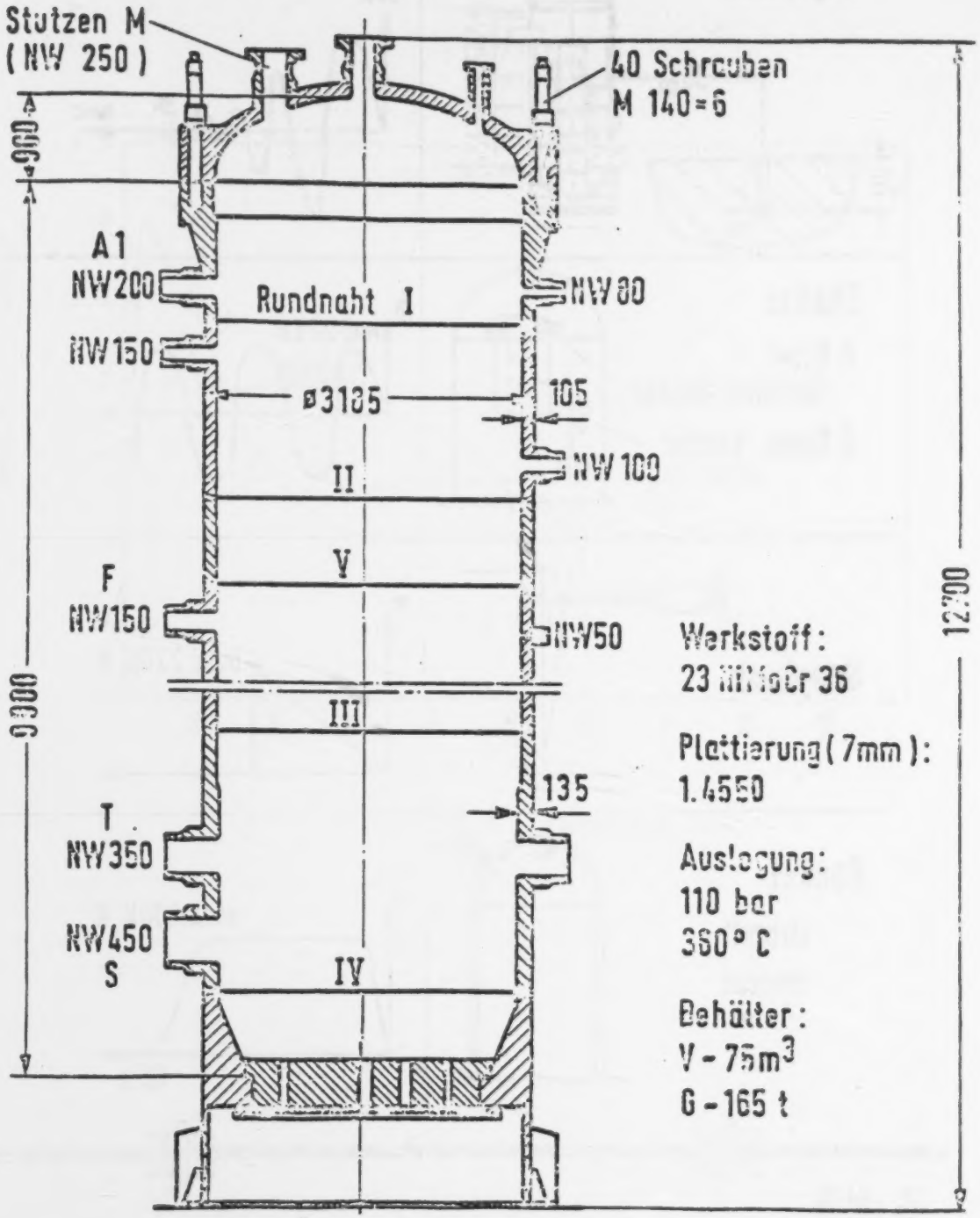
Bild: 1



(REAKTORBUILDING)

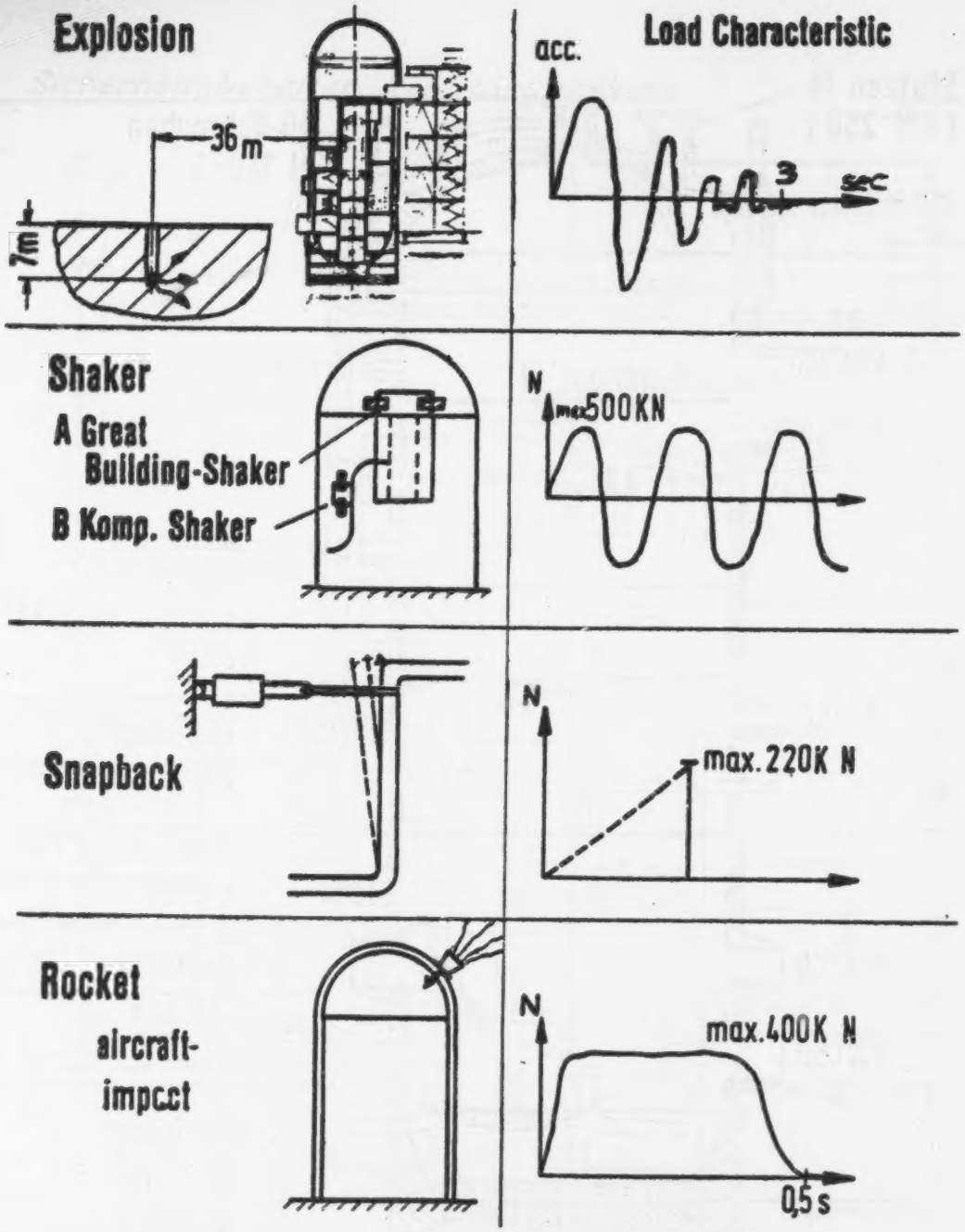
REAKTORDRUCKGEBÄUDE LÄNGSSCHNITTE

BILD 2



REAKTORDRUCKBEHÄLTER (RPV)

BILD 3



HDR-Erthquake-Tests: Exitation-methods

Bild: 4

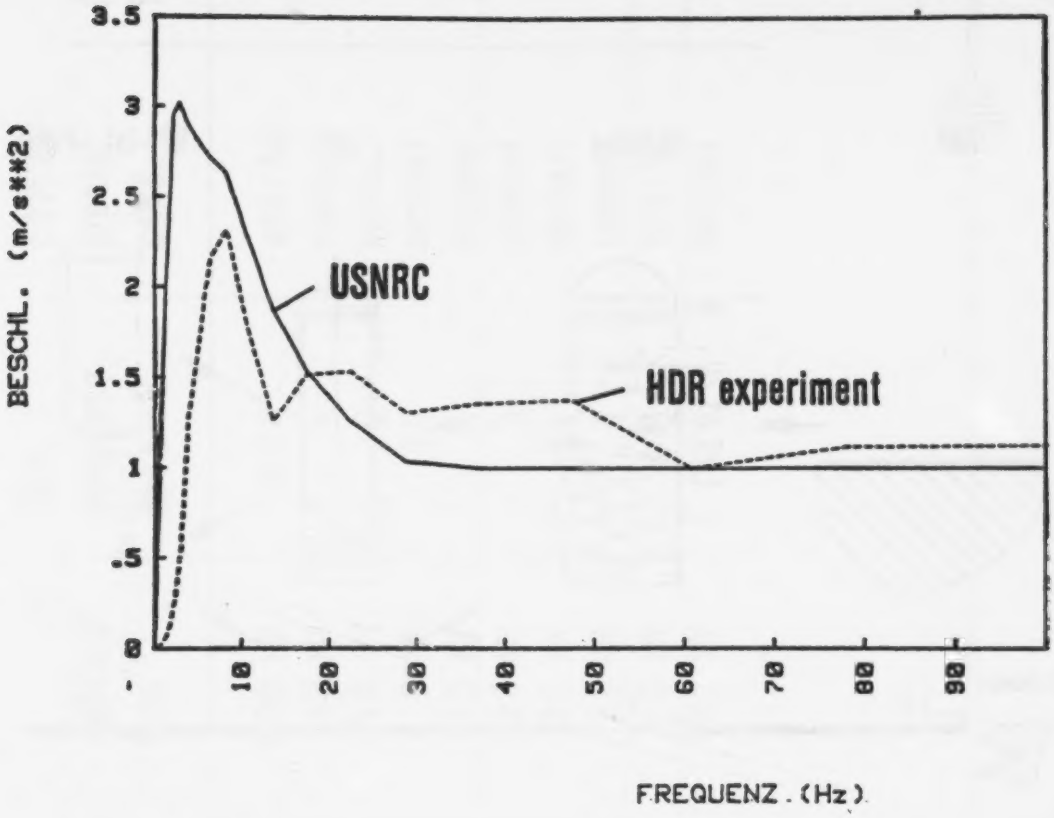
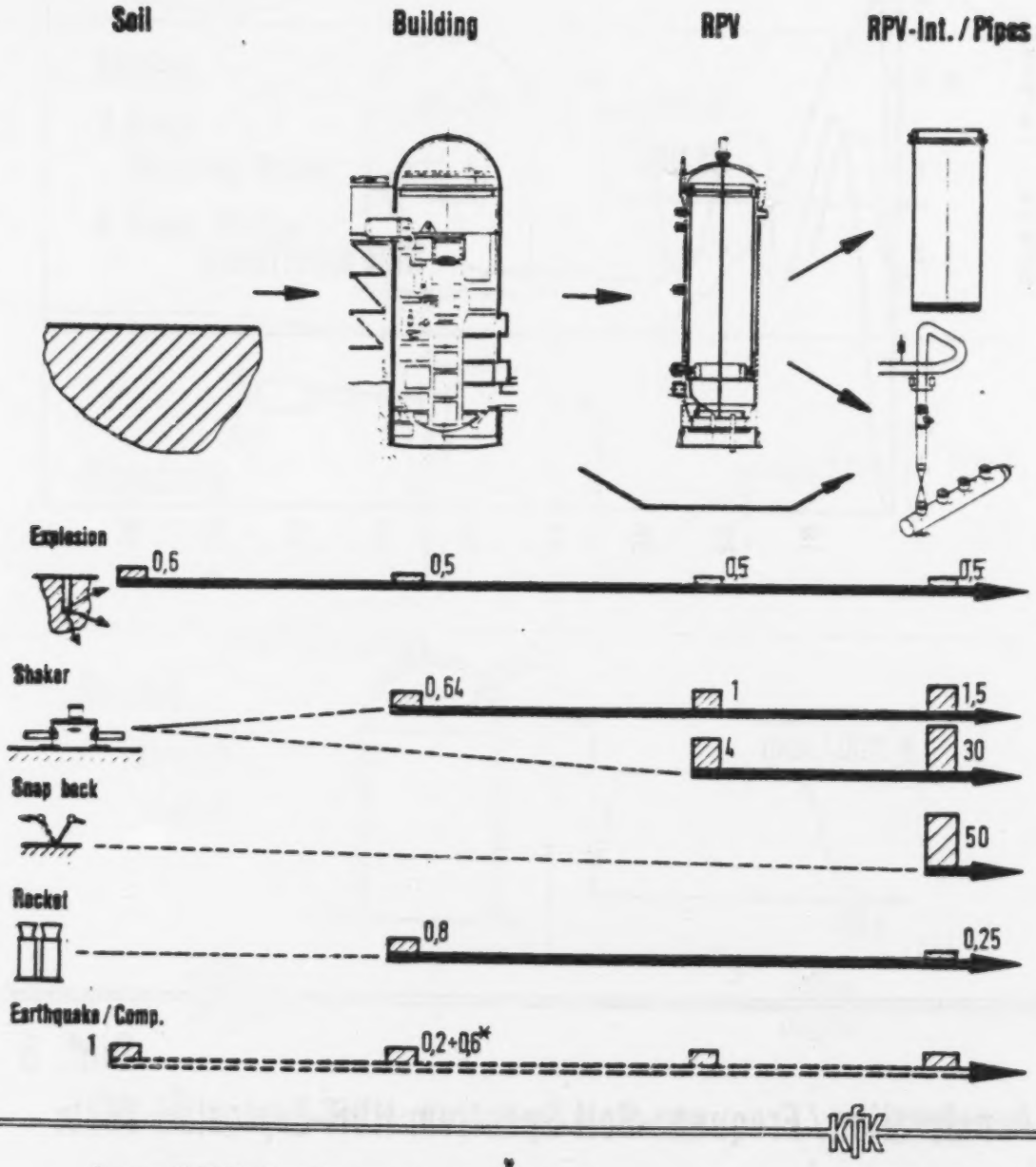


Bild: 5

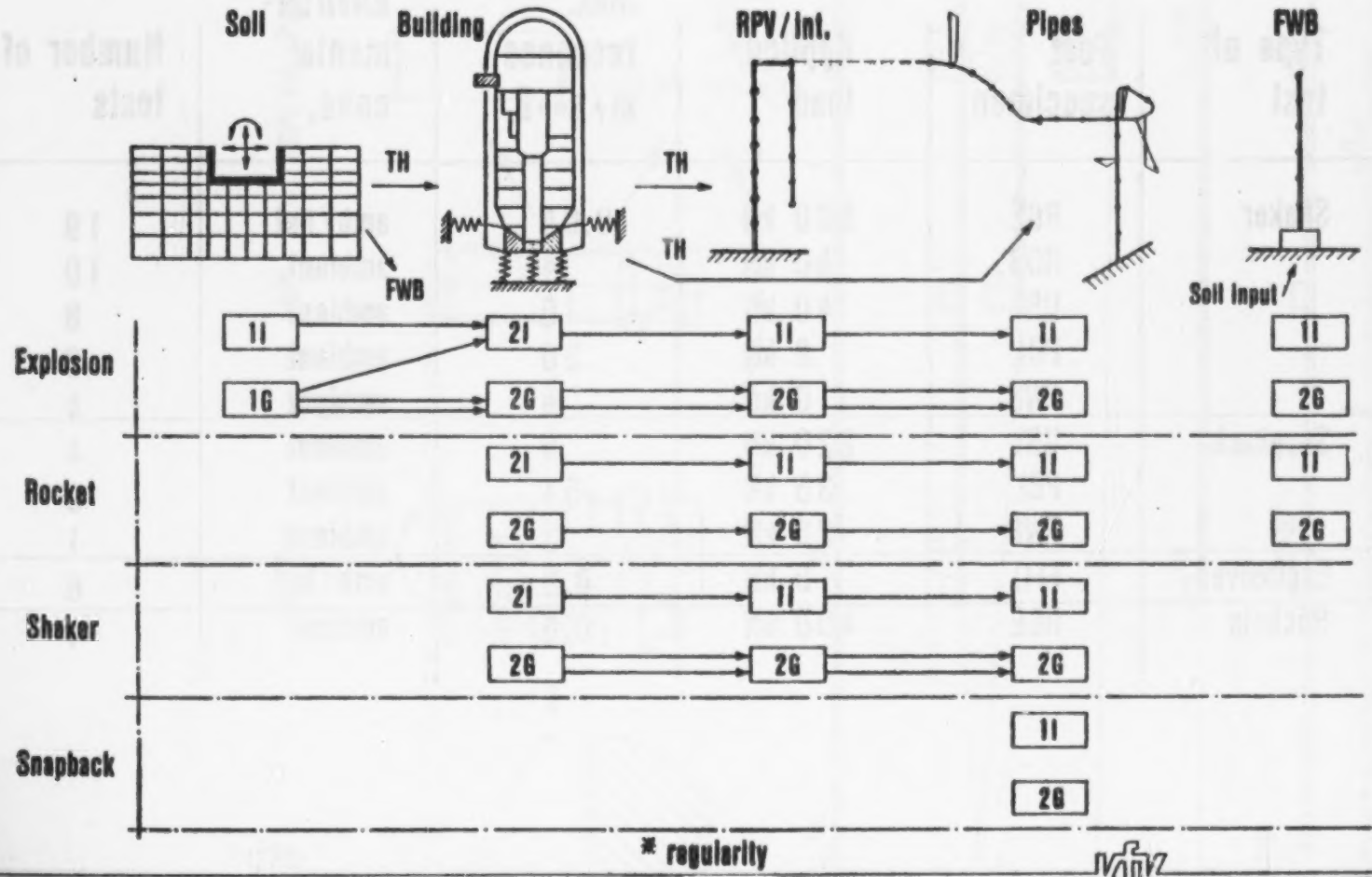
Acceleration / Frequenz-Soil Spectrum HDR-Explosion Tests



Survey HDR-Earthquake experiments (accel. m/s^2)

Bild: 6

Type of test	Test specimen	Applied load	Max. response m/s**2	Environ-mental cond.	Number of tests
Shaker	RGE	500 kN	0.64	amb/hot	19
	RDB	40 kN	4	ambient	10
	URL	40 kN	16	ambient	8
	PDL	8 kN	30	ambient	6
	FWB	10 kN	4	ambient	4
Snapback	URL	220 kN	9	ambient	4
	PDL	35 kN	51	ambient	5
	FWB	23 kN	3	ambient	1
Explosive	FFD	5 kN	0.6	amb/hot	6
Rockets	RGE	400 kN	0.6	ambient	4



Survey: Pretest Calculations HDR-Earthquake tests
(I: industrie group, G: * group, 1 / 2 sep.calc.)

Bild: 8

Design Calculation

— estimate load levels for tests
— optimize transducer locations

Safety Calculations

- safety of important HDR-components
— safety of nearby power plants

Pre-Calculations

— natural frequencies and mode shapes
— dynamic response for sinusoidal and transient loads

Blind Post-Calculations

— input values from test (blast tests only)

Post-Calculations

— parameter identification
— improved computer codes

Institution	Component	Code	
Industry group			
Battelle	FFD	DSIM	FD, non-linear
Hochtief	RGE (SSI)	BAUBOW	FE, linear
KWU	RGE	MESY	FE, linear
	RDB	ANSYS	FE, linear
	URL, PDL	KWUROHR	FE, part.-non-lin.
	FWB	STARDYNE	FE, linear
Regulatory group			
ESI	FFD	HEMP	FD, non-linear
	RGE (SSI)	PAM-GEOM	FE, linear
König	RGE	FRQDOM	FE, linear
GRS	RGE, FWB	ASKA	FE, linear
SDF	RDB, URL, PDL	ASKA	FE, linear
TÜV	RDB, URL, PDL	SAP	FE, part.-non-lin.
SDK	FWB	SAN-RO	FE, linear

Components

	building / Soil	RPV / FUT	Pipings
1. Damping	●	●	●
2. Modeling (discr., idealisation)	● +	● +	● +
2a Coupling / decoupling		●	●
2b load propagation	●		
3. Boundry cond. Supports			●
3a Stiffness		●	●
3b Soil / structure interaction	●		
4. Stresses		●	●
5. Calcul. methods linear, non-l., quasi-l.			●
5a Approximate methods	●		●

× Subj. low level E.

KfK

Main Subjects: HDR-earth quake invest. **Bild: 11**

		Planned Priority *	RGE	RDB+KS	RL	FWB
Determination of Parameters of the real structure	**Parameters:**					
	- Damping	1	⊗	⊗	⊗	⊗
	- Stiffness	2	X	X	X	X
	- Mass	2	X	X	X	X
	Boundary Conditions:					
	- Coupling	2	X	X	X	
	- Support	1			⊗	
	- Gaps, Restraints	5		X	X	
	Dependences:					
	- Load Dependence of Parameters	2	X	X	X	X
	- Dependence of Parameters from environment	2		X	X	
Description of the real Structure by a model	**Modelling Technics:**					
	- Kind of model (Beam, Shell, Contin.)	3**	X	X	X	X
	- Coupling/Decoupling	2	X	X	X	
	- Degree of Discretisation	3**	X	X	X	X
	- Damping	1	X	X	⊗	⊗
	Boundary Conditions:					
	- Support, Gaps, Restraints	1			⊗	X
	- Soil-Structure Interaction	1	⊗			
	- Load Propagation	1		⊗	⊗	
	- Influence of Water	2		X		X
Possible better Procedure for the description of the real vibration behavior	**Calculations with experimentally verified models:**					
	- natural vibration behavior	2	X	X	X	X
	- forced vibration behavior	2	⊗	⊗	⊗	⊗
	- strain, stress	2	(X)	X	X	X
	Consideration of Nonlinearities					
	- nonlinear calculation	3			⊗	
	- quasilinear calculation	2	X		X	
	Simplified Methods	1				

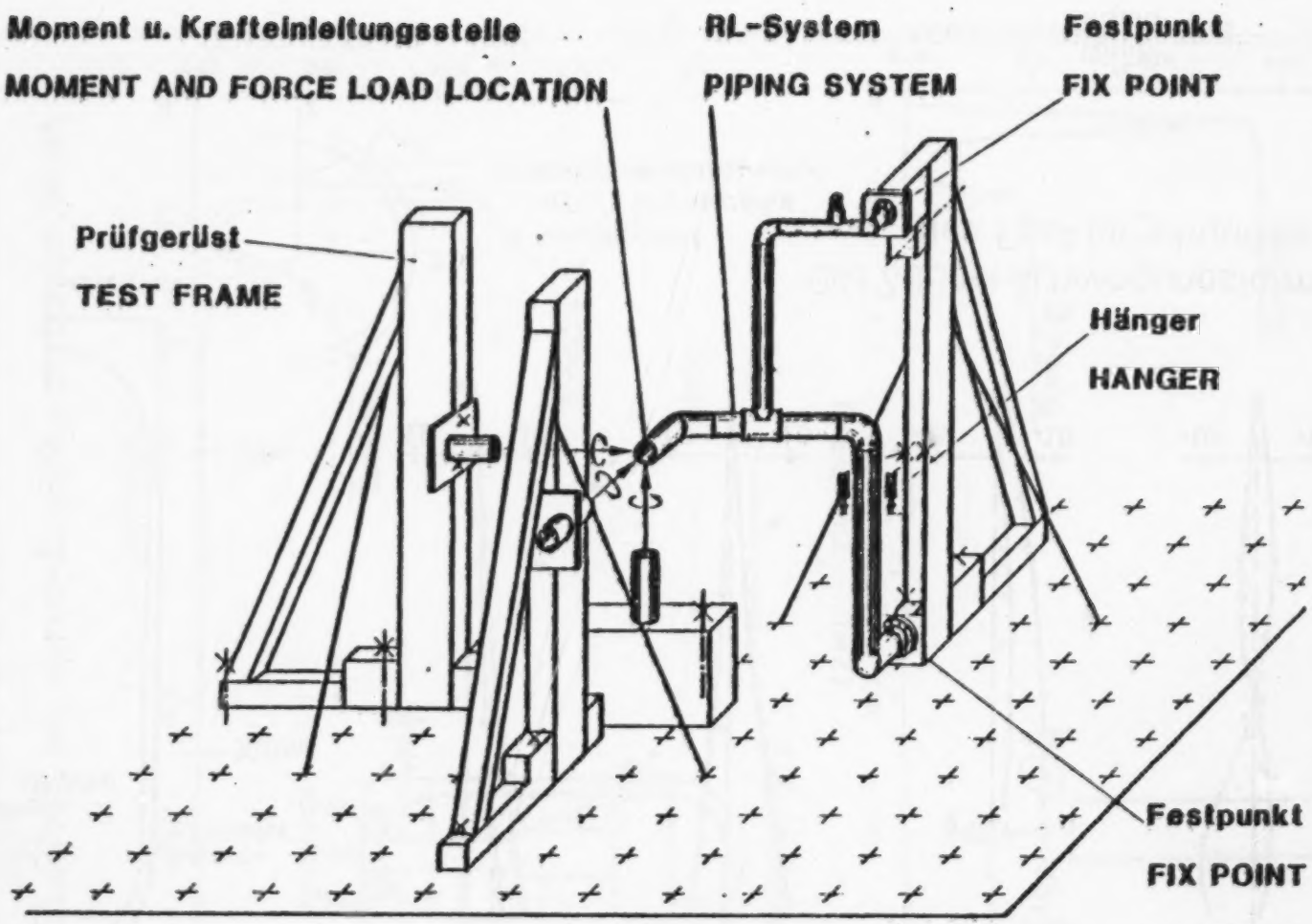
X Data available
⊗ extended studies planned
* 1 = highest priority
** extended studies during low level phase

RGE : Reactorbuilding with soil
RDB+KS : Reactor pressure vessel with core barrel
RL : Pipes
FWB : Flood water tank

KfK

BILD 11A

MAIN POINTS OF INVESTIGATION DURING EARTHQUAKESTUDIES AT HDR



PIPING TEST RIG

BILD 12

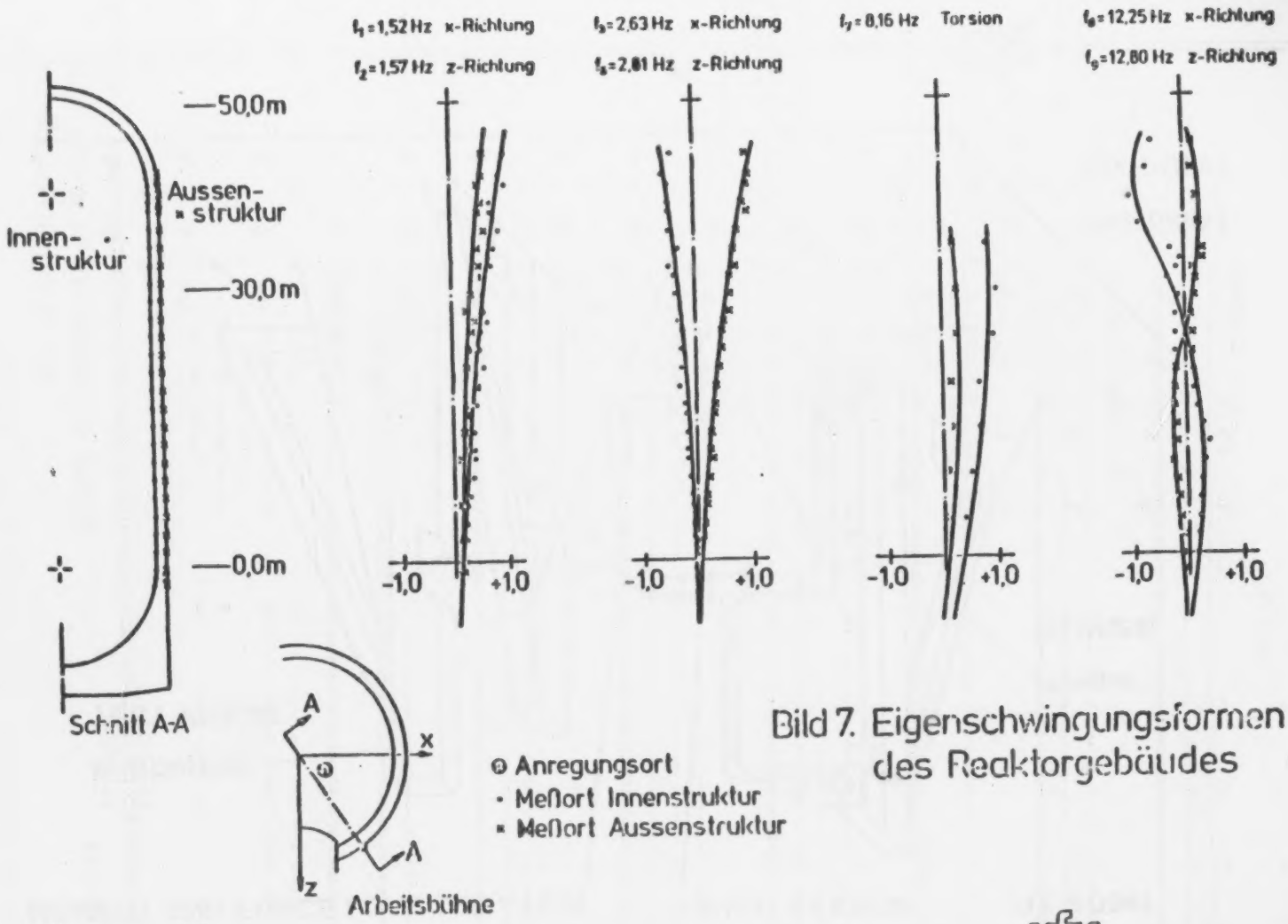
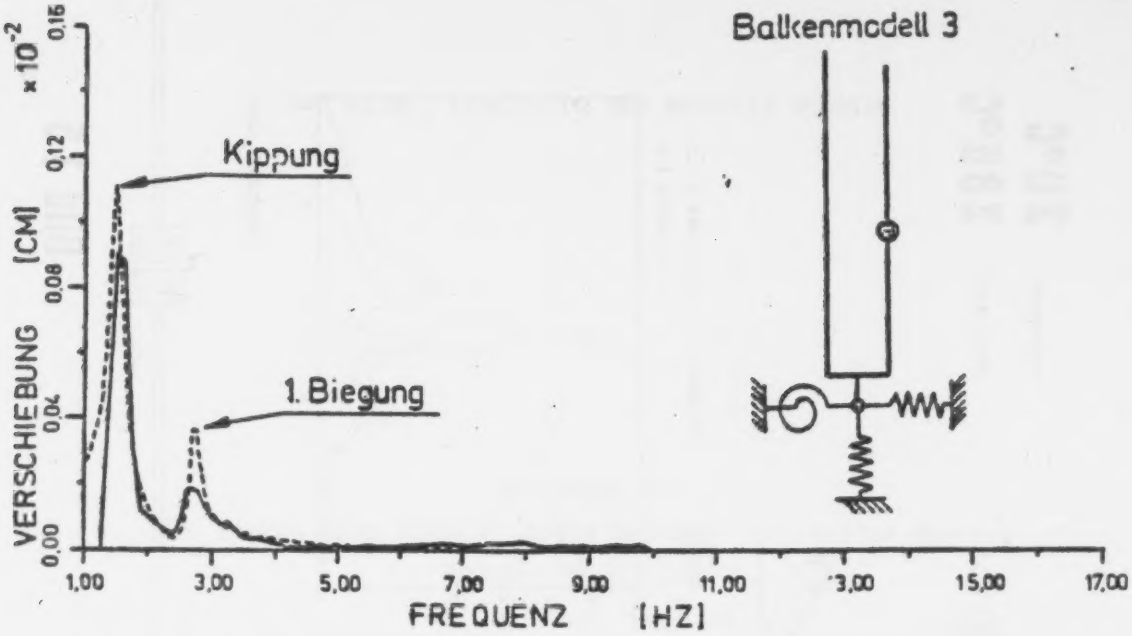
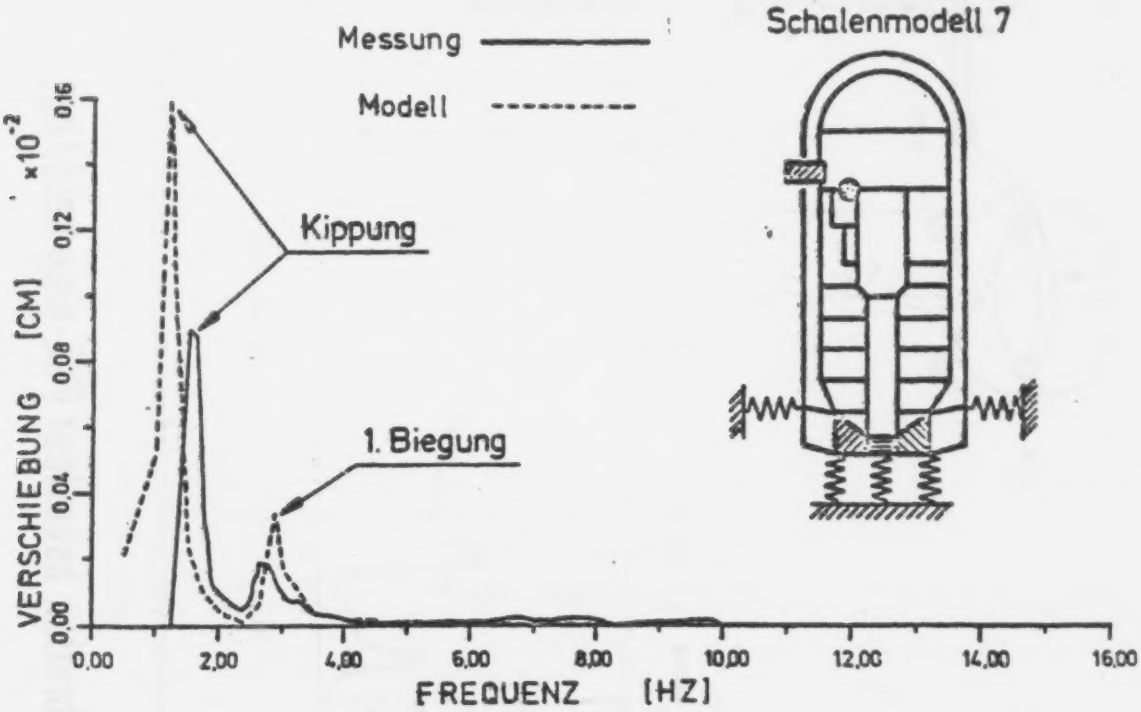


Bild 7. Eigenschwingungsformen des Reaktorgebäudes



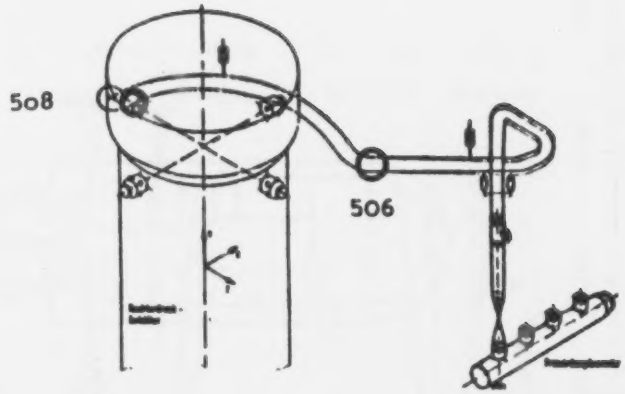


Verschiebung im Reaktorgebäude bei Shakeranregung

KTK

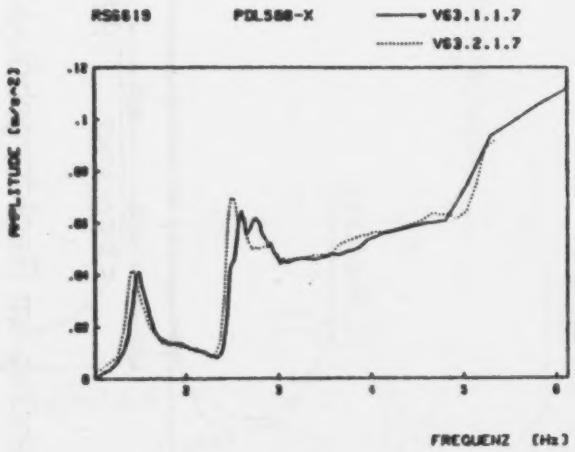
HDR-BUILDING DISPLACEMENT

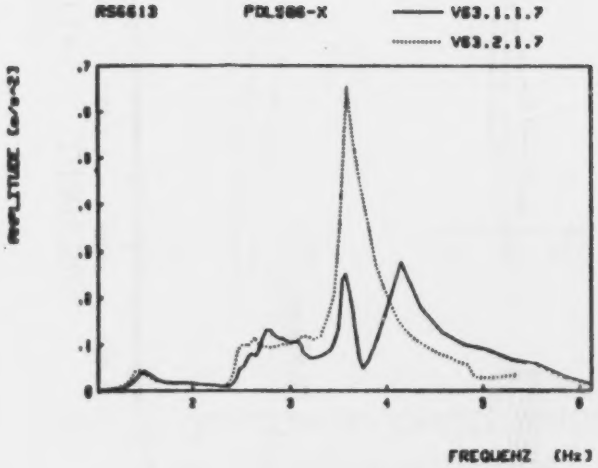
BILD 14



20°C

285°C





kfk

Vibration behaviour PDL-pipe cold / hot

Bild 15

HDR SEISMIC SIMULATION AND ANALYSIS PROGRAM

Presented at
The Eighth Water Reactor Safety Research Information Meeting
October 27-31, 1980
Gaithersburg, Maryland

R. G. Rahl
EG&G Idaho, Inc.

Idaho National Engineering Laboratory
Idaho Falls, Idaho 83415

HDR SEISMIC SIMULATION AND ANALYSIS PROGRAM

R. G. Rahl
EG&G Idaho, Inc.

In order to better understand the response of nuclear power plant piping systems subjected to ground shaking, NRC's Office of Nuclear Regulatory Research, Division of Reactor Safety Research, has initiated a cooperative effort with the Federal Republic of Germany with respect to the Heissdampfreaktor (HDR) testing program. The HDR is a decommissioned reactor being used for seismic testing.

EG&G Idaho, is supporting the NRC by making "blind" predictions of the response of the HDR recirculation loop piping to explosive exitation of the HDR containment. These predictions were made using a nonlinear transient time history computer analysis with measured input accelerations (at system anchor points) obtained from ANCO Engineers of California.

After release of blind predictions EG&G Idaho was furnished with measured accelerations at points on the piping system. These accelerations were used to generate response spectra and compared to corresponding spectra which were generated using predicted accelerations.

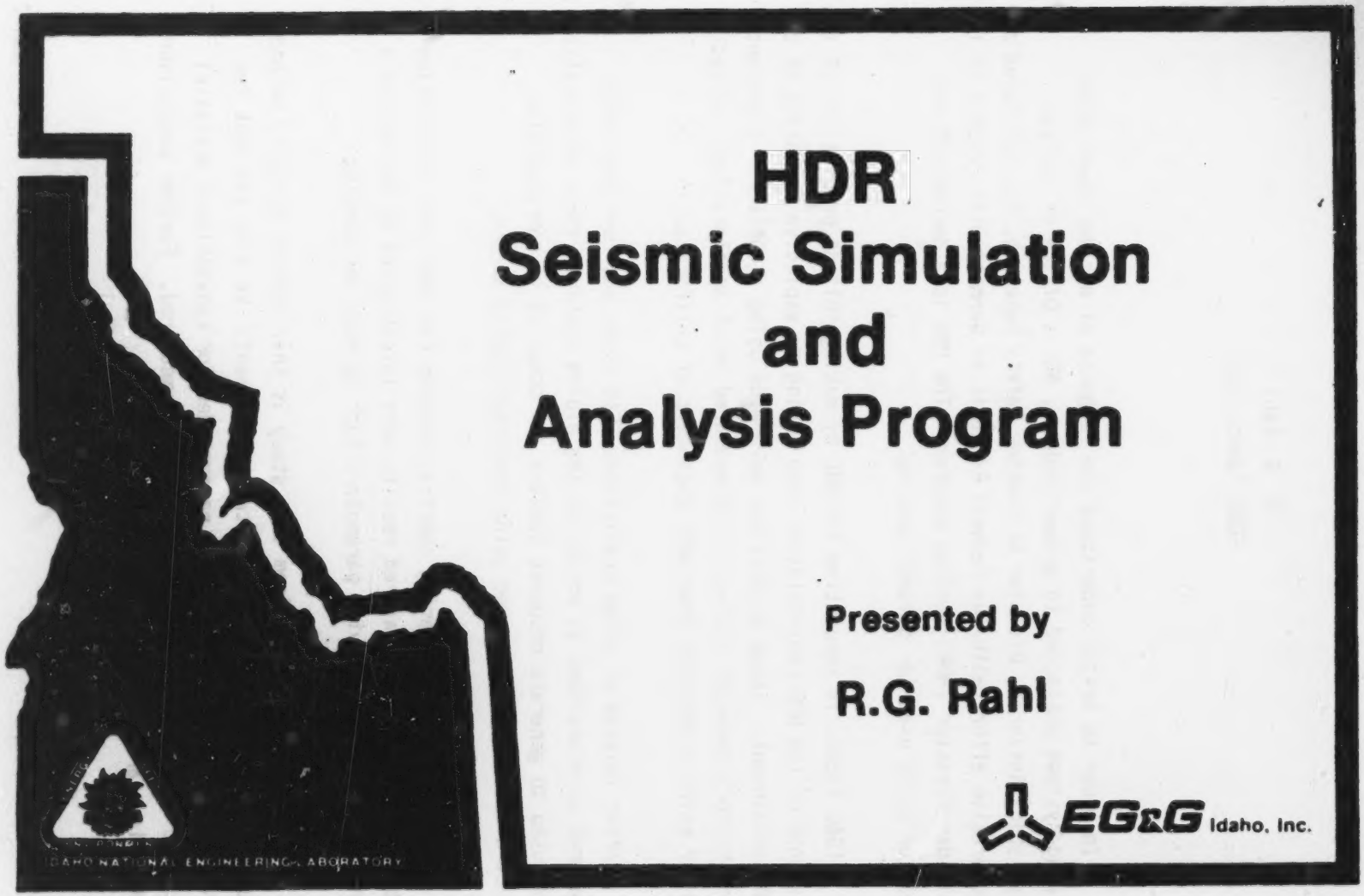
In addition to comparing spectra, reasons for some poor correlation between predicted and measured results were investigated by performing a linear analysis and varying parameters such as mass and damping.

One conclusion drawn from this study is that current analysis methods predict conservative results for this experiment. We also saw that for many cases 7% damping (versus 1% recommended for conventional analysis) produced results more nearly equal to those measured. Further comparison evaluations are to be continued in FY-81.

HDR Seismic Simulation and Analysis Program

Presented by

R.G. Rahl

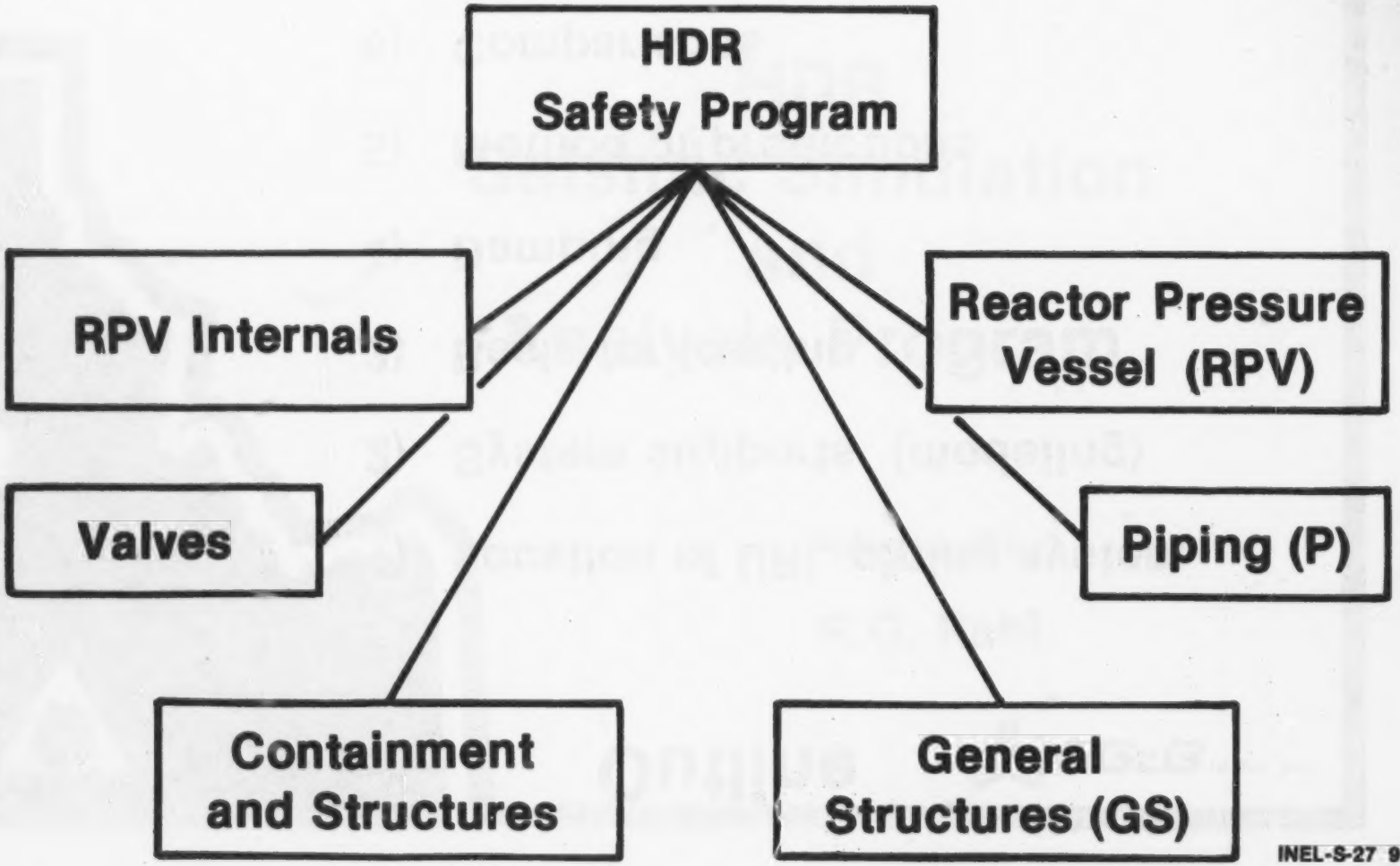


Outline

1) Location of URL piping system
2) System supports (modeling)
3) Basis for loading
4) Damping
5) Method of predictions
6) Comparisons
7) Conclusions

INEL-S-28 219

Component Categories



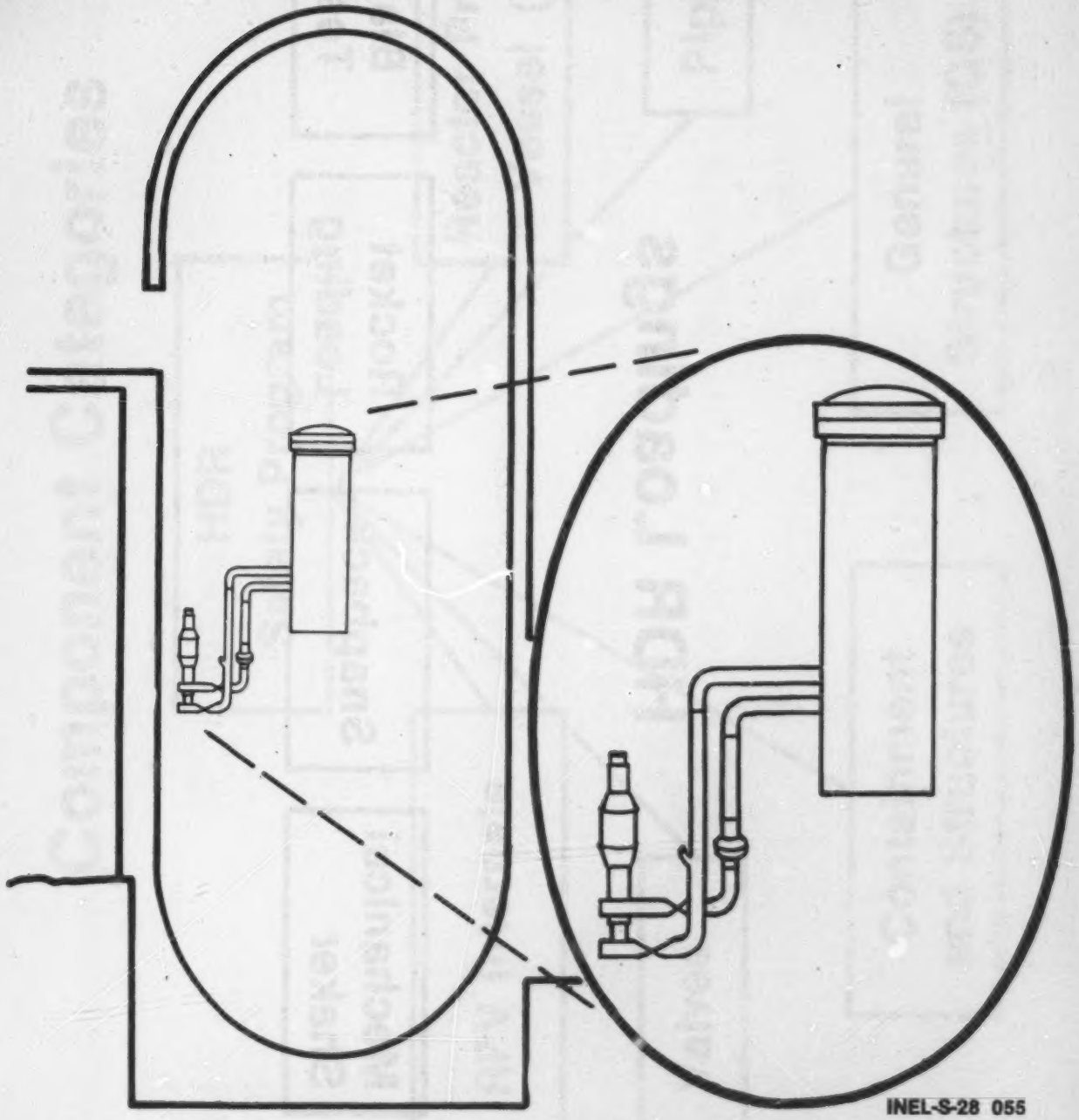
INEL-S-27 906

HDR Loadings

Mechanical Shaker	Snapback	Rocket Loading	Blast Testing

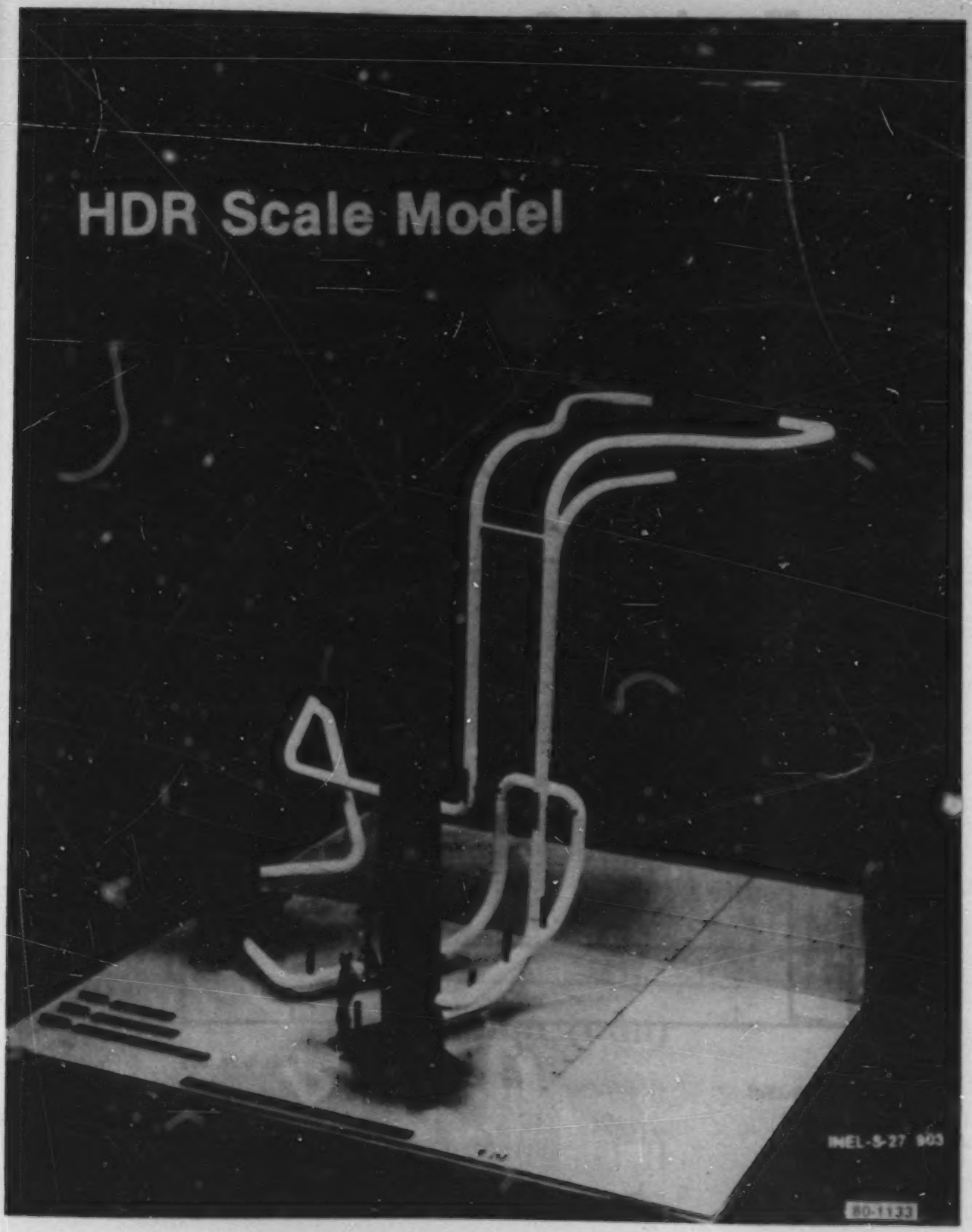
INEL-S-27 904

X-Section Showing URL System

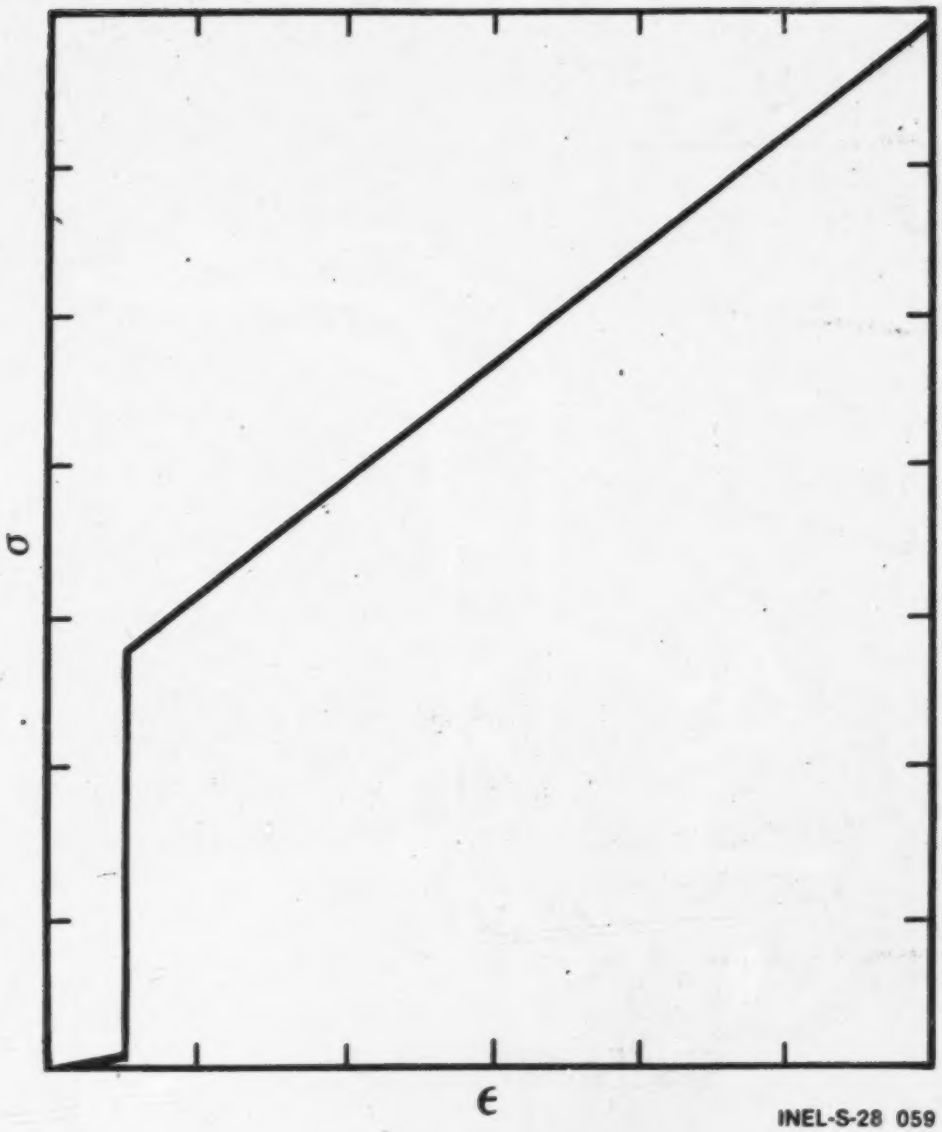


INEL-S-28 055

HDR Scale Model

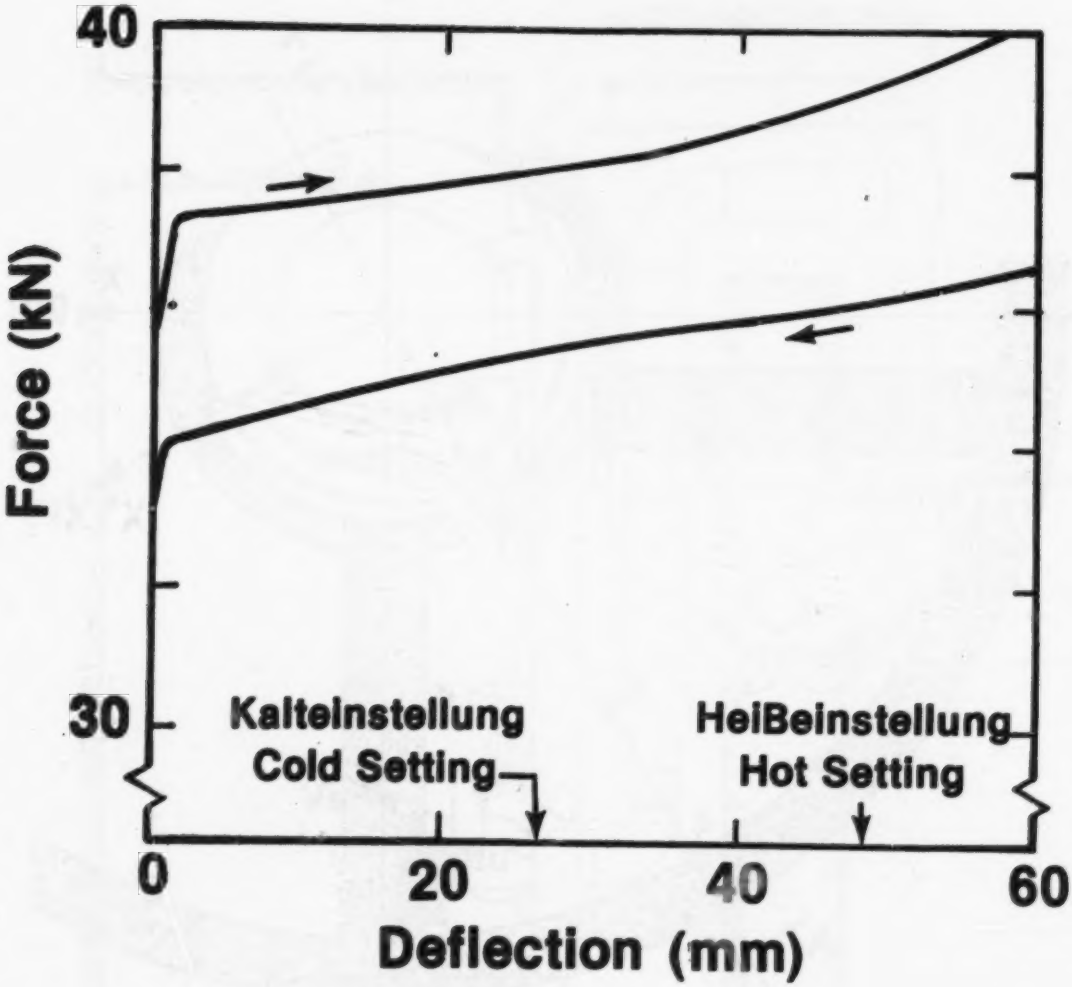


Typical Sway Brace Stress-Strain Curve



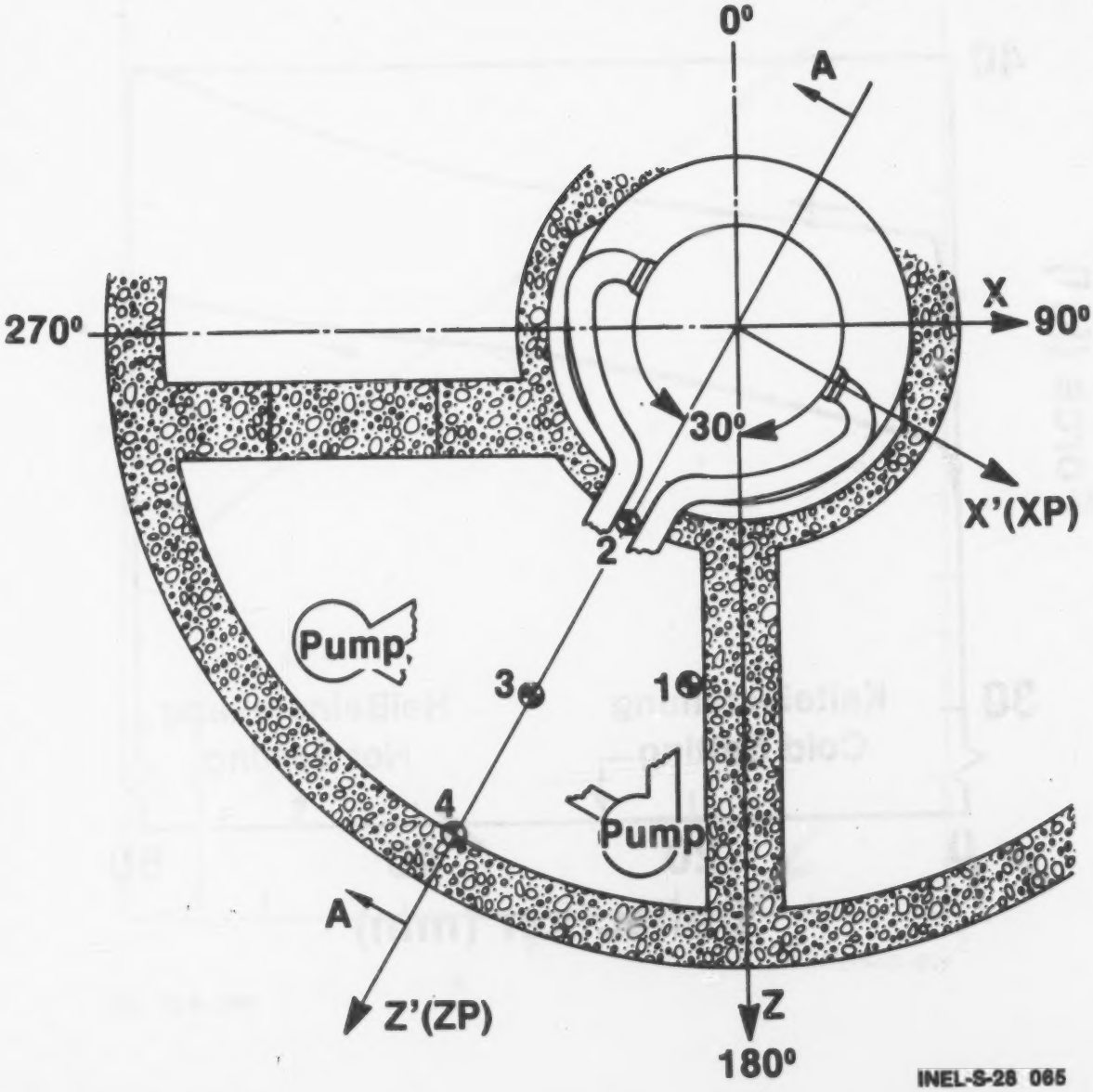
INEL-S-28 059

Typical Force-DEFL Curve for Constant Force Hanger

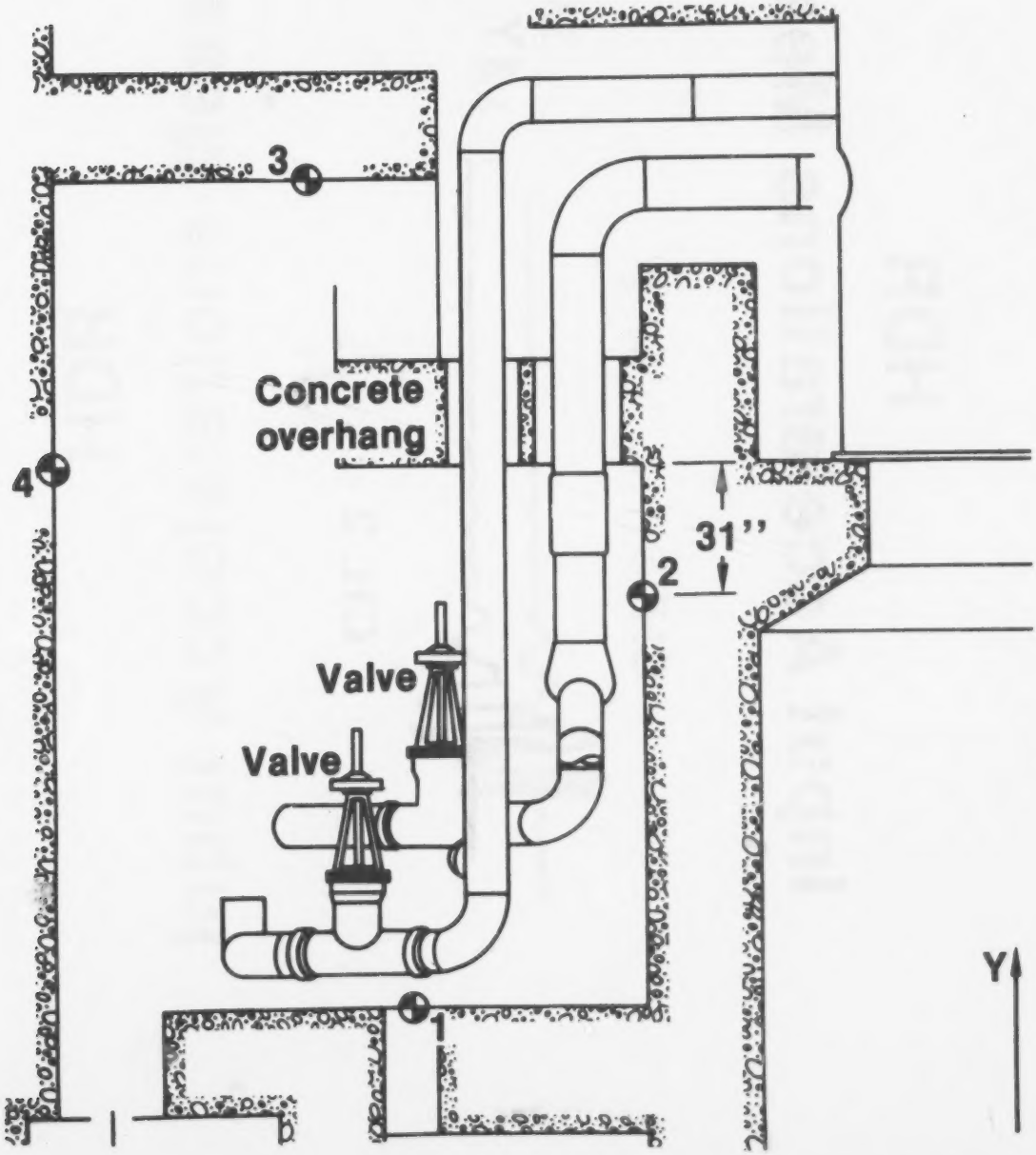


INEL-S-28 058

URL Loop Location and Instrumentation Points Plan View



URL Loop Location and Instrumentation Points Elevation



INEL-S-28 084

HDR
Input Accelerations Measured

1Y

CH. 2

INEL-S-28 066

HDR
Input Accelerations Measured

1Y

2Y

CH. 2 CH. 5

INEL-S-28 066

HDR
Input Accelerations Measured

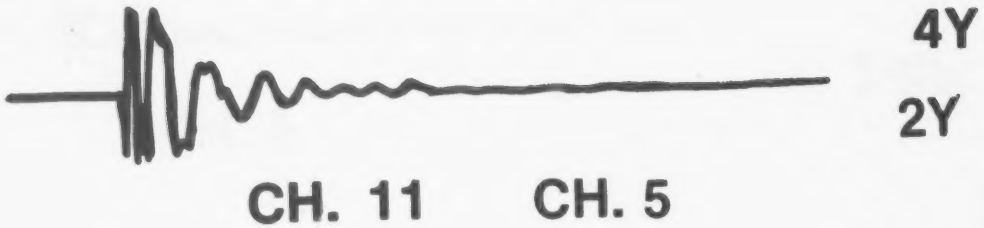
1Y

3Y

CH. 2 CH. 8

INEL-S-28 066

HDR
Input Accelerations Measured



INEL-S-28 0t

HDR
Input Accelerations Measured

2-ZP

CH. 6

INEL-S-28 066

HDR
Input Accelerations Measured

2-ZP

CH. 6

INEL-S-28 066

HDR
Input Accelerations Measured

CH. 4

2-XP

INEL-S-28 066

HDR
Input Accelerations Measured

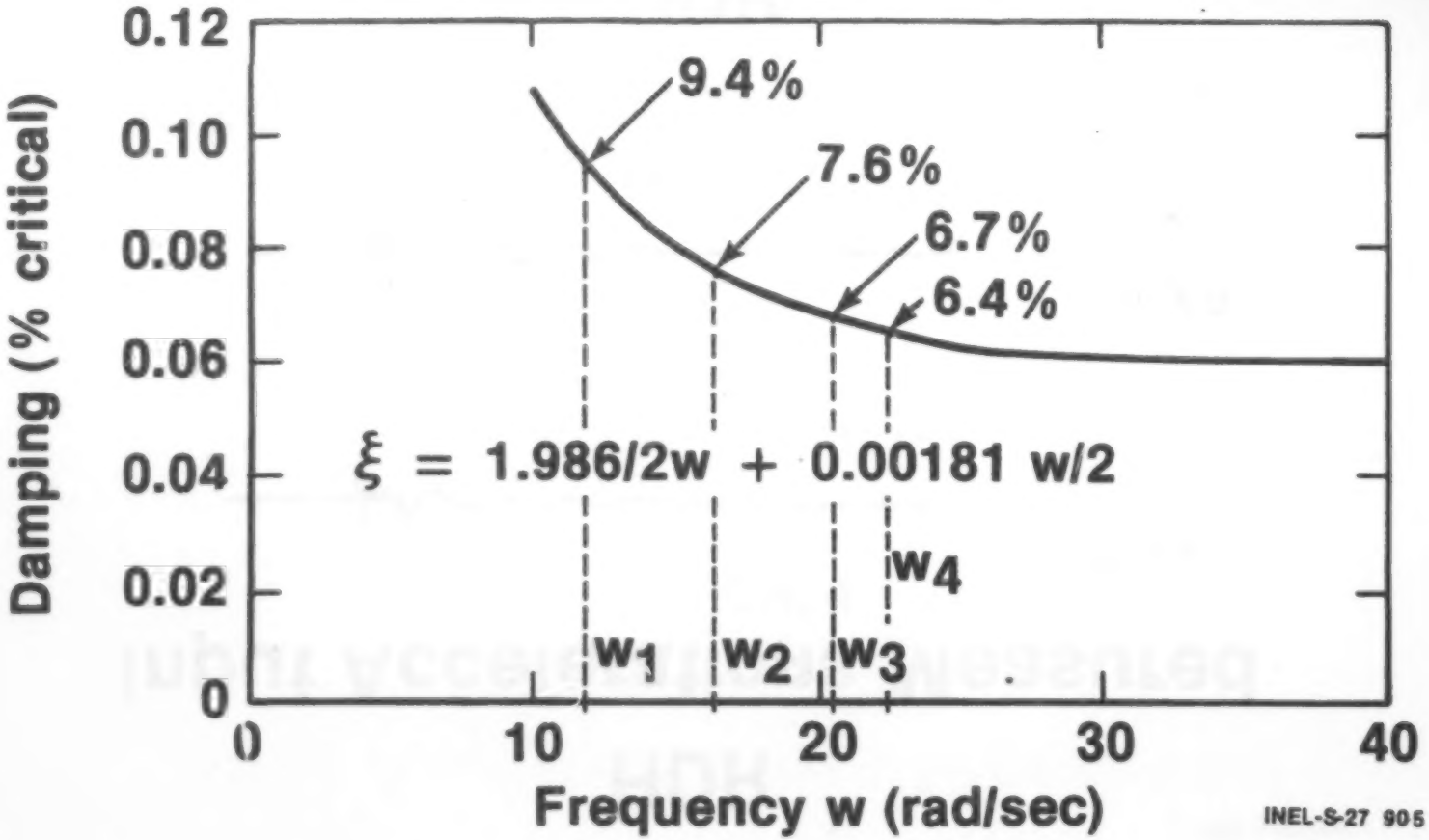
CH. 10 CH 4

4-XP

2-XP

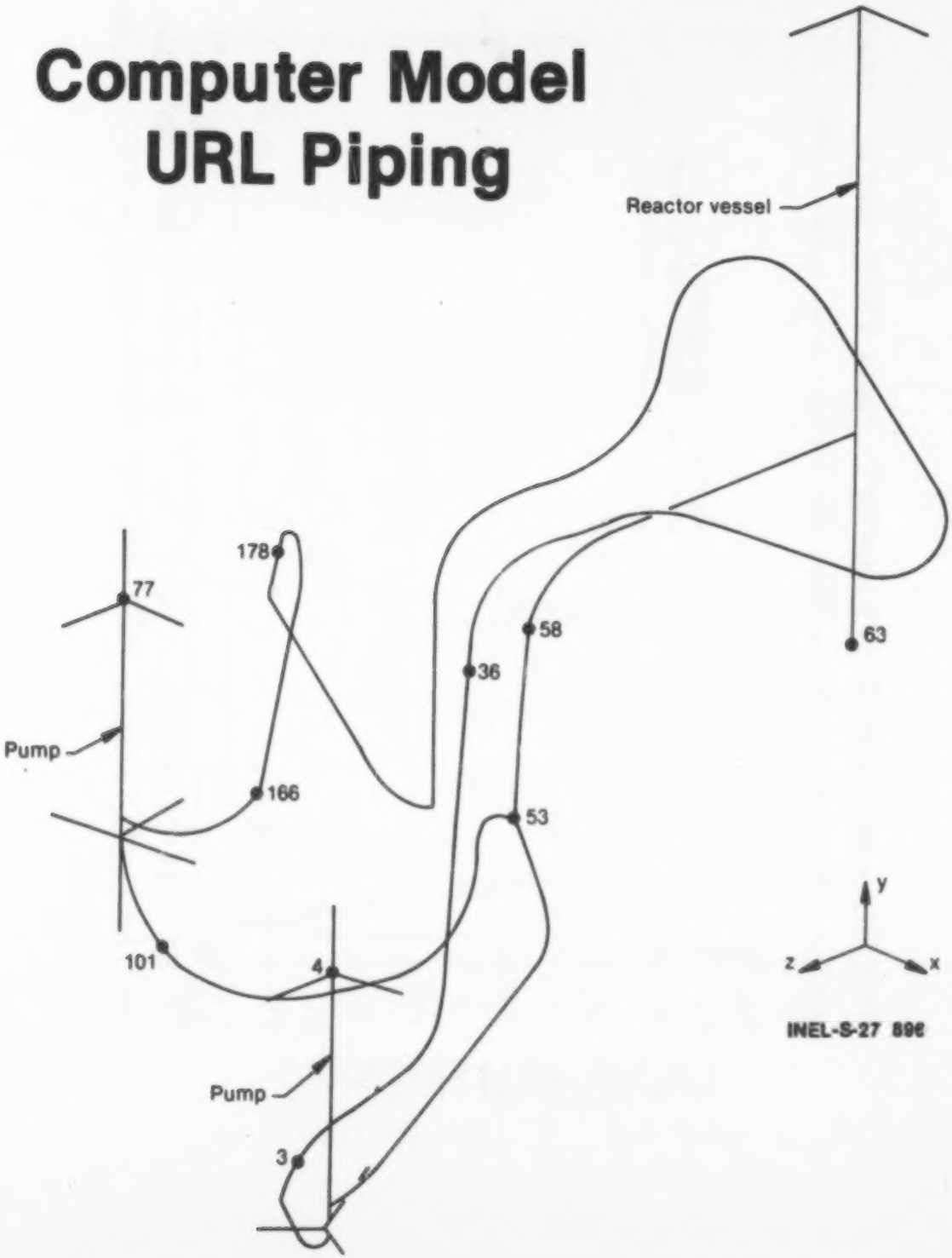
INEL-S-28 066

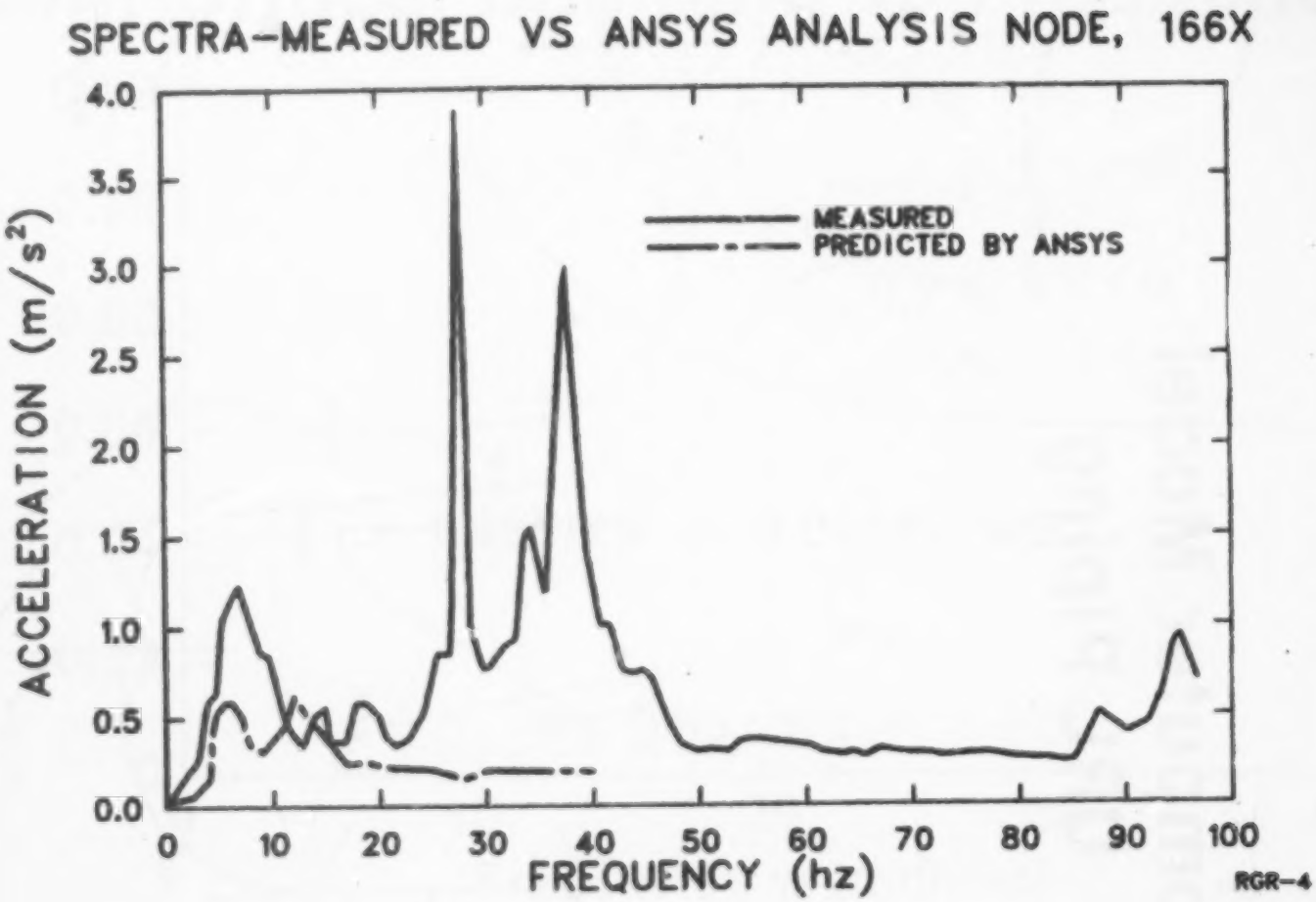
Percent Critical Damping vs Frequency

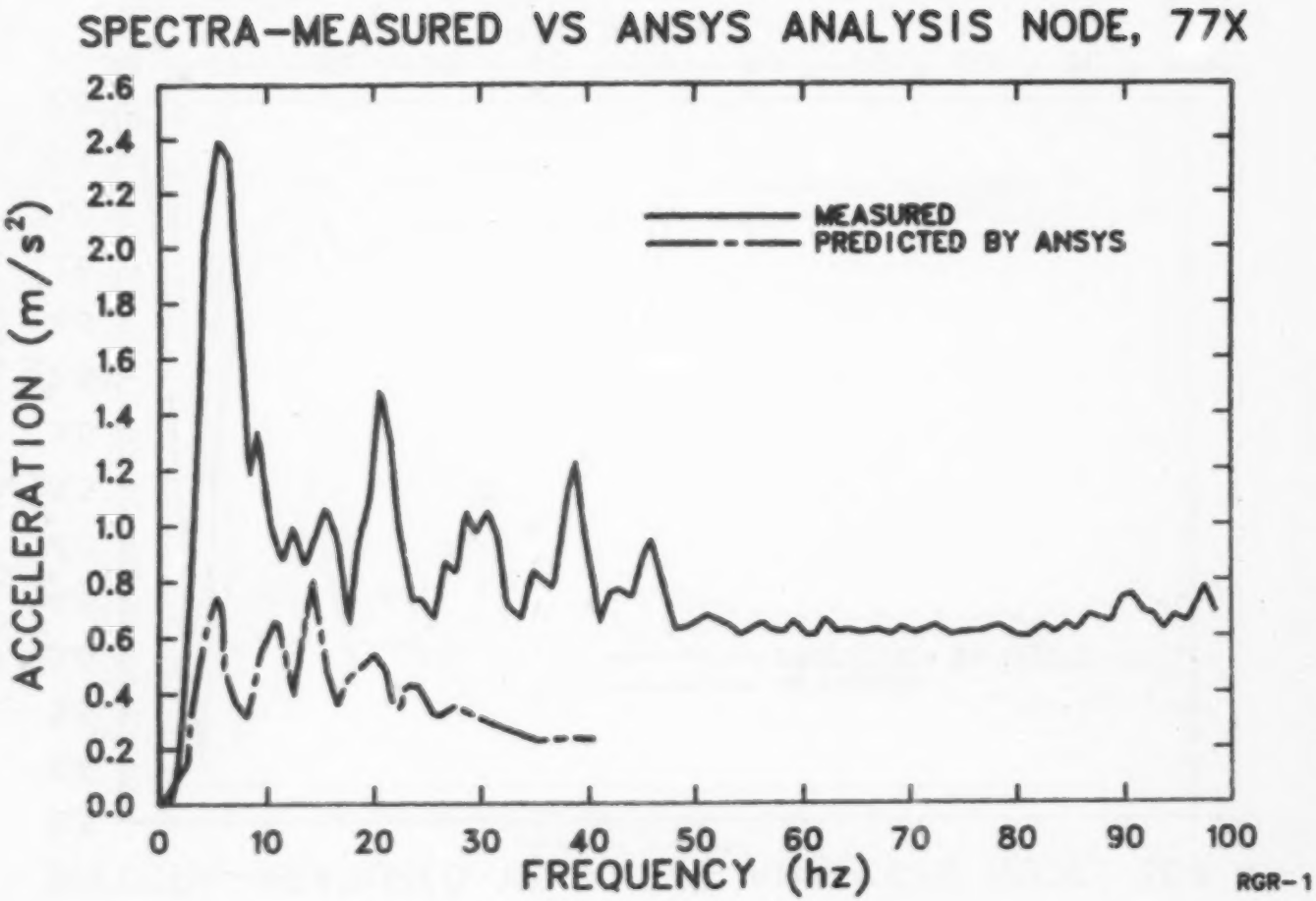


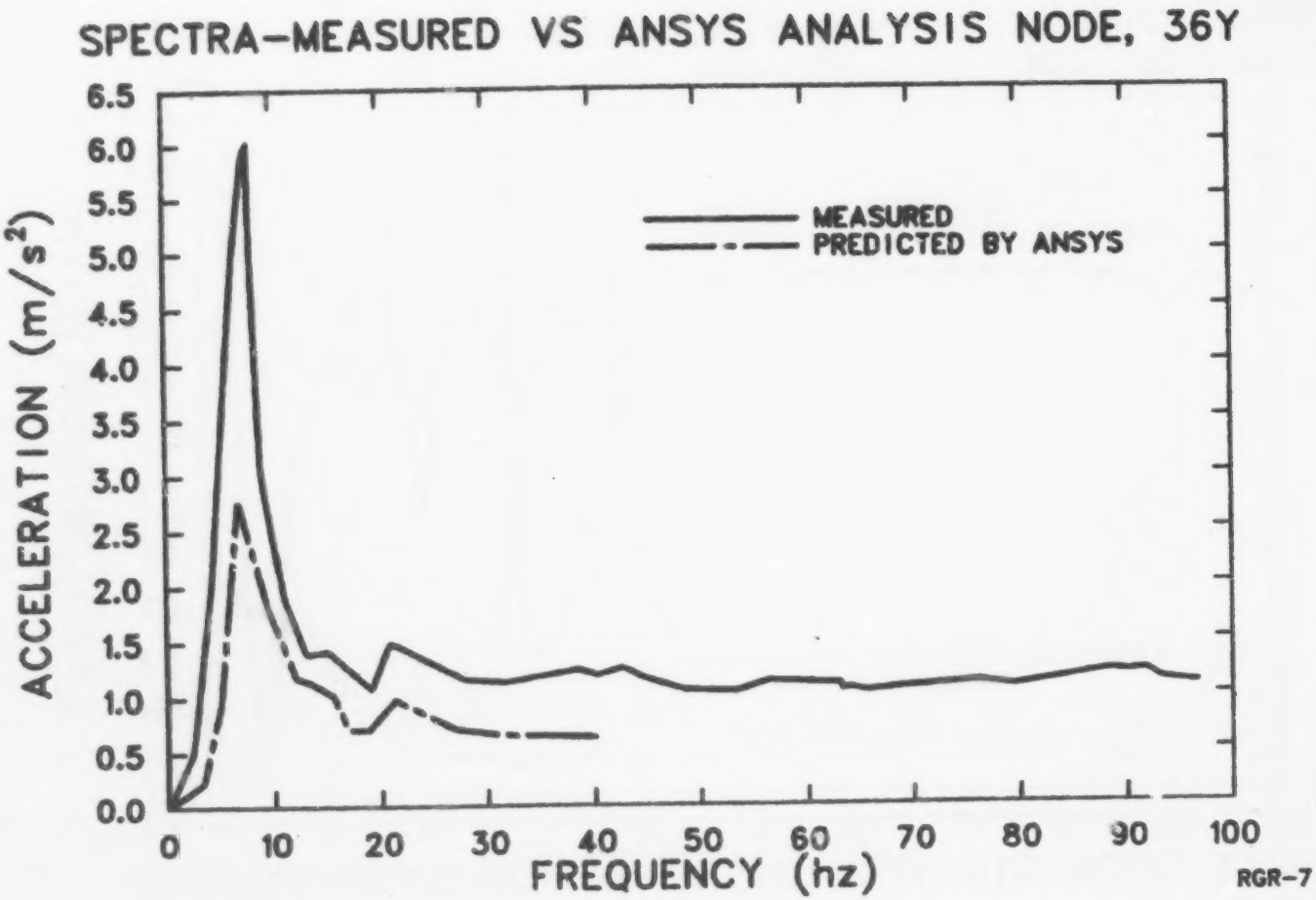
INEL-S-27 905

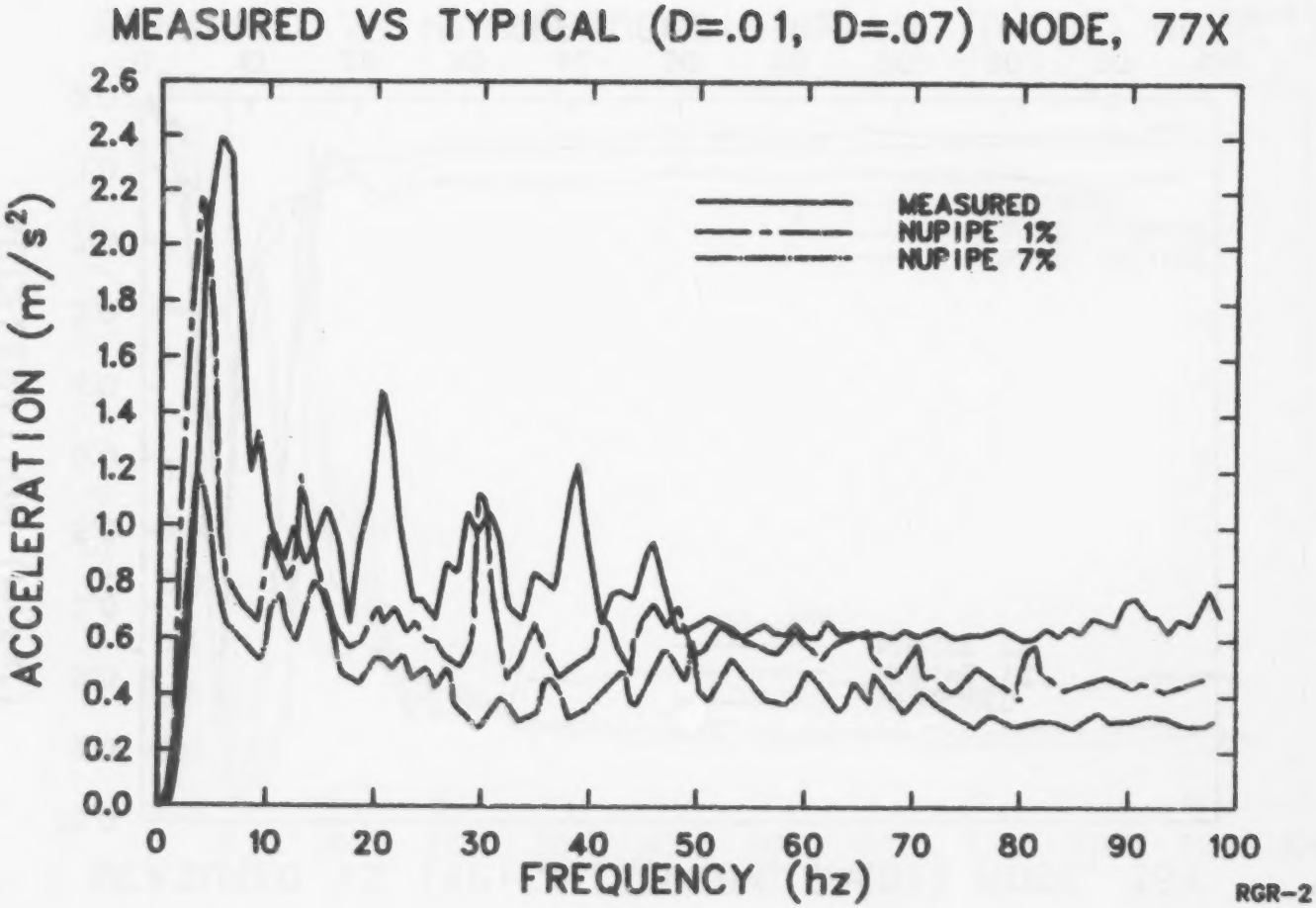
Computer Model URL Piping

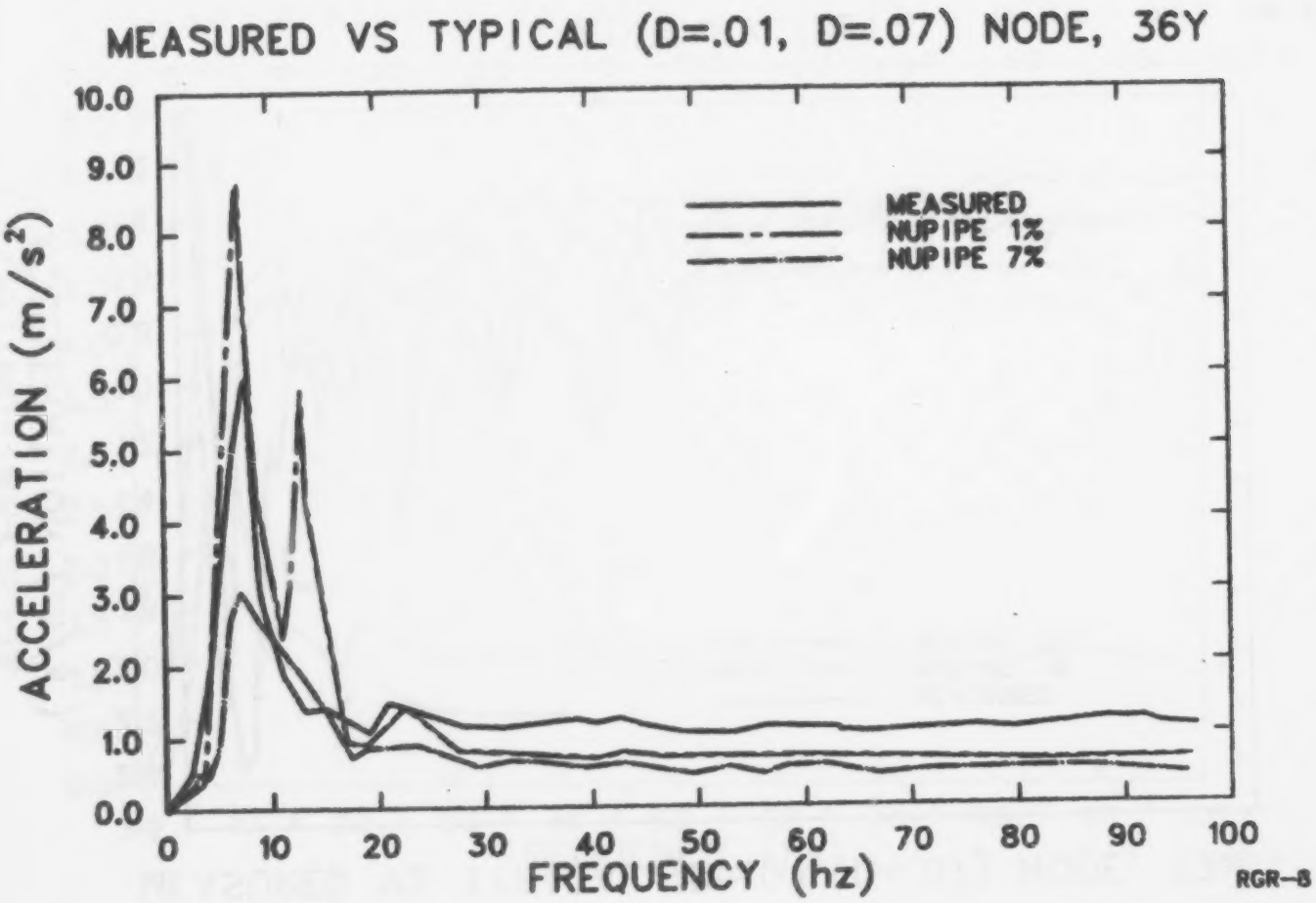


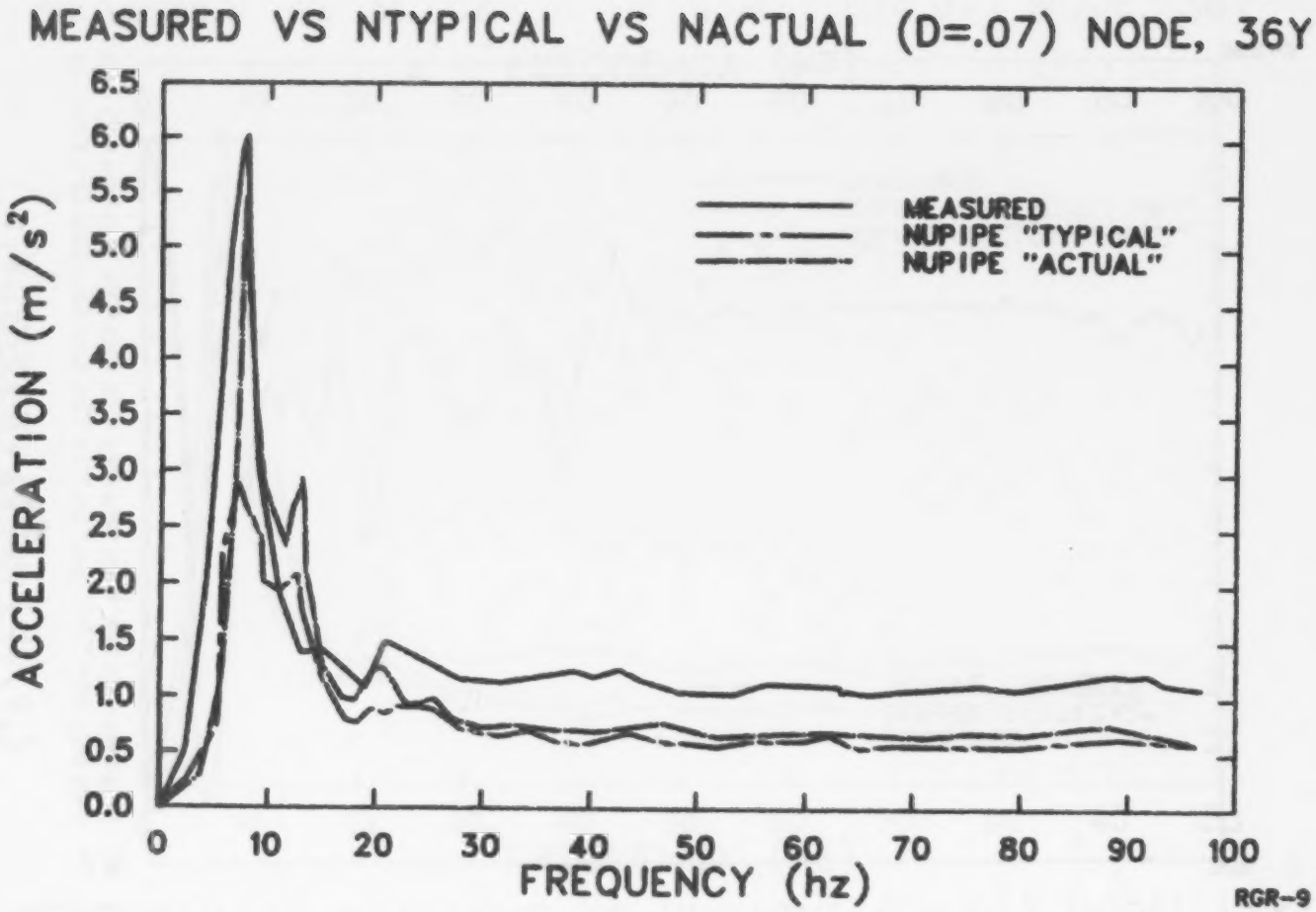

SPECTRA-MEASURED VS ANSYS ANALYSIS NODE, 166X
MEASURED
PREDICTED BY ANSYS
ACCELERATION (m/s²)
4.0
3.5
3.0
2.5
2.0
1.5
1.0
0.5
0.0
0
10
20
30
40
50
60
70
80
90
100
FREQUENCY (hz)
RGR-4

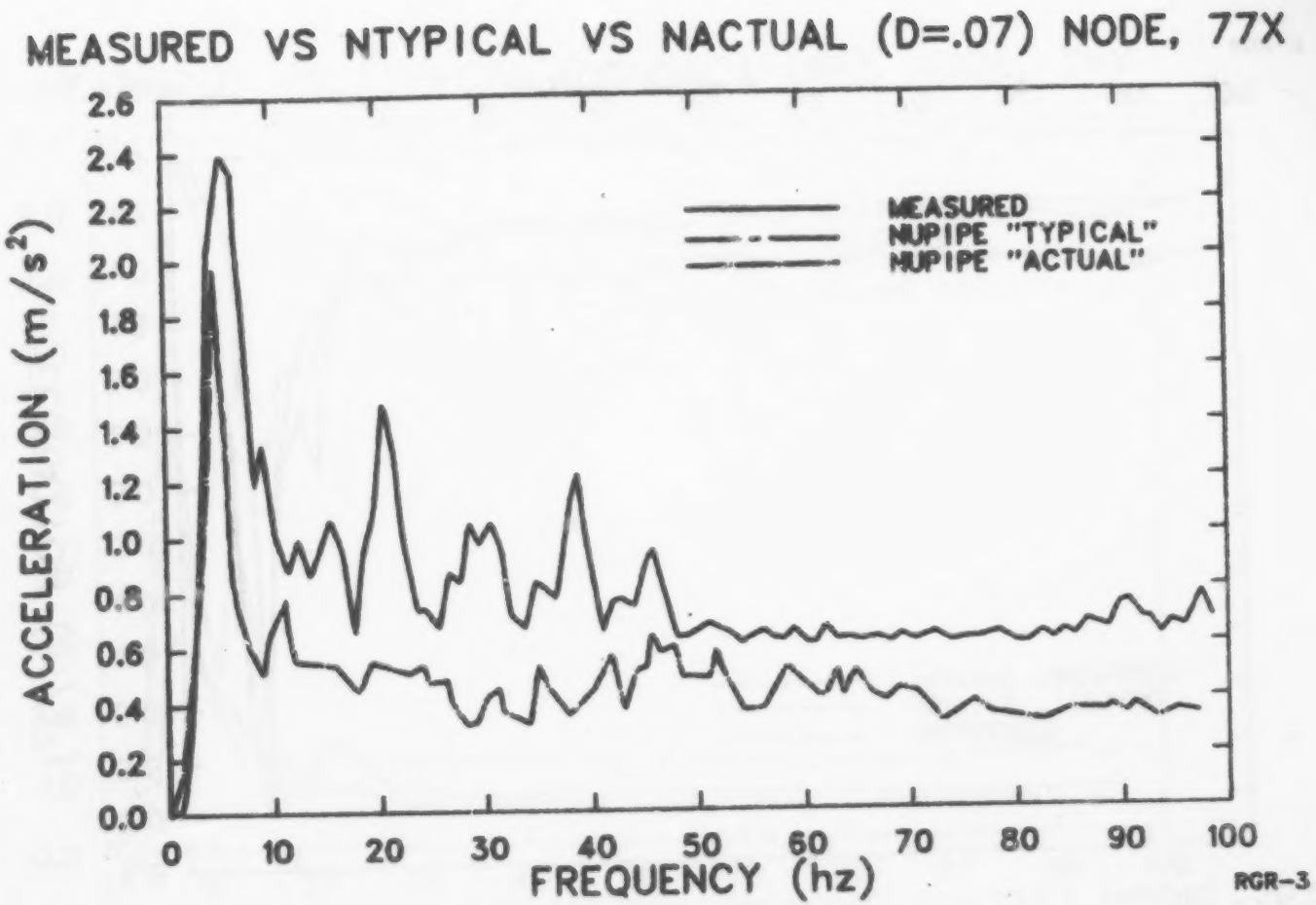

SPECTRA-MEASURED VS ANSYS ANALYSIS NODE, 77X
MEASURED
PREDICTED BY ANSYS
ACCELERATION (m/s²)
2.6
2.4
2.2
2.0
1.8
1.6
1.4
1.2
1.0
0.8
0.6
0.4
0.2
0.0
0
10
20
30
40
50
60
70
80
90
100
FREQUENCY (hz)
RGR-1

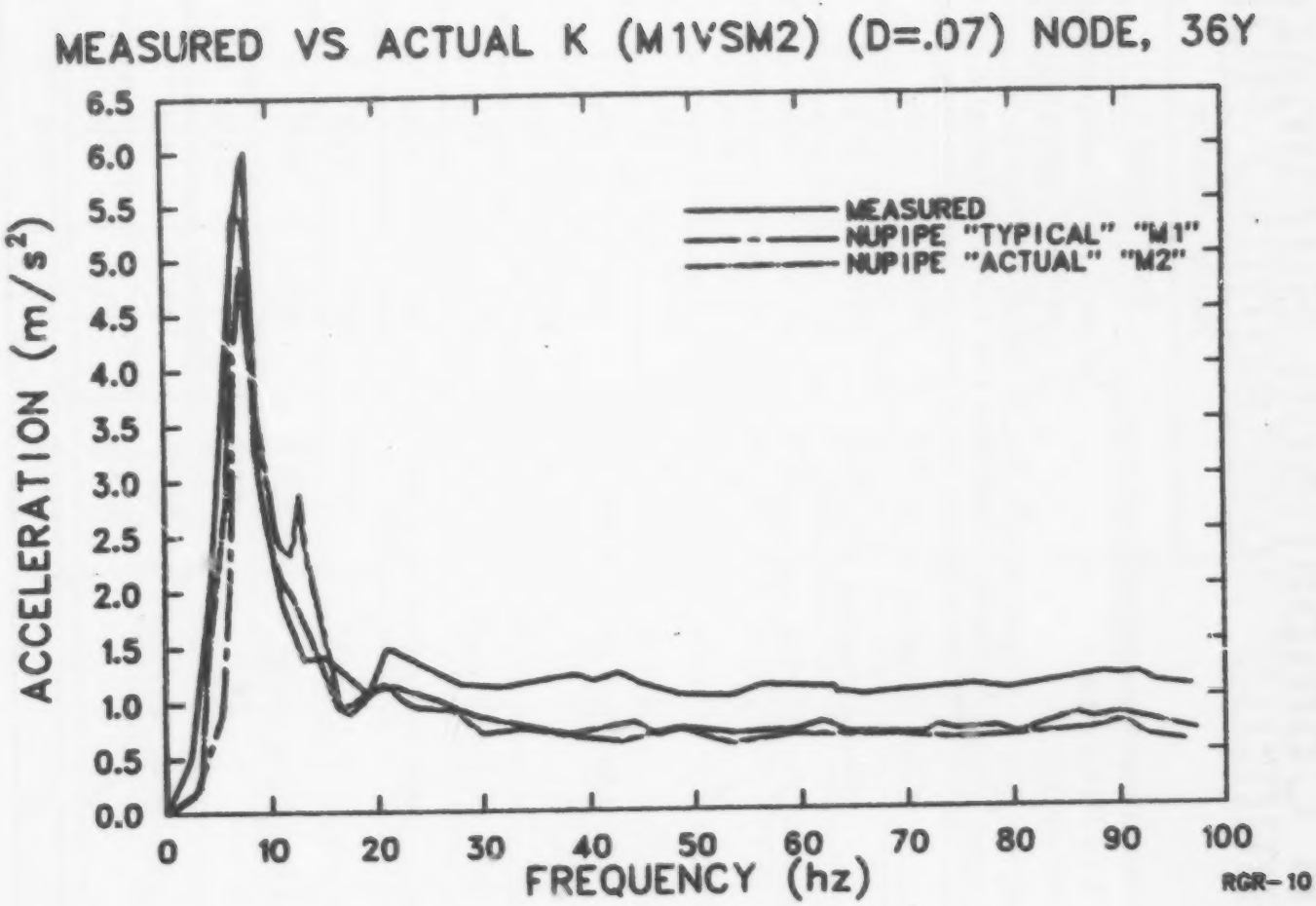

SPECTRA-MEASURED VS ANSYS ANALYSIS NODE, 36Y
ACCELERATION (m/s²)
6.5
6.0
5.5
5.0
4.5
4.0
3.5
3.0
2.5
2.0
1.5
1.0
0.5
0.0
0
10
20
30
40
50
60
70
80
90
100
FREQUENCY (hz)
MEASURED
PREDICTED BY ANSYS
RGR-7


MEASURED VS TYPICAL (D=.01, D=.07) NODE, 77X
ACCELERATION (m/s²)
2.6
2.4
2.2
2.0
1.8
1.6
1.4
1.2
1.0
0.8
0.6
0.4
0.2
0.0
0
10
20
30
40
50
60
70
80
90
100
FREQUENCY (hz)
MEASURED
NUPIPE 1%
NUPIPE 7%
RGR-2


MEASURED VS TYPICAL (D=.01, D=.07) NODE, 36Y
ACCELERATION (m/s²)
10.0
9.0
8.0
7.0
6.0
5.0
4.0
3.0
2.0
1.0
0.0
0
10
20
30
40
50
60
70
80
90
100
FREQUENCY (hz)
MEASURED
NUPIPE 1%
NUPIPE 7%
RGR-8


MEASURED VS NTYPICAL VS NACTUAL (D=.07) NODE, 36Y
ACCELERATION (m/s²)
6.5
6.0
5.5
5.0
4.5
4.0
3.5
3.0
2.5
2.0
1.5
1.0
0.5
0.0
0
10
20
30
40
50
60
70
80
90
100
FREQUENCY (hz)
MEASURED
NUPIPE "TYPICAL"
NUPIPE "ACTUAL"
RGR-9


MEASURED VS NTYPICAL VS NACTUAL (D=.07) NODE, 77X
ACCELERATION (m/s²)
2.6
2.4
2.2
2.0
1.8
1.6
1.4
1.2
1.0
0.8
0.6
0.4
0.2
0.0
0
10
20
30
40
50
60
70
80
90
100
FREQUENCY (hz)
MEASURED
NUPIPE "TYPICAL"
NUPIPE "ACTUAL"
RGR-3


MEASURED VS ACTUAL K (M1VSM2) (D=.07) NODE, 36Y
ACCELERATION (m/s²)
6.5
6.0
5.5
5.0
4.5
4.0
3.5
3.0
2.5
2.0
1.5
1.0
0.5
0.0
0
10
20
30
40
50
60
70
80
90
100
FREQUENCY (hz)
MEASURED
NUPIPE "TYPICAL" "M1"
NUPIPE "ACTUAL" "M2"
RGR-10

HDR Comparison Summary

				Nupipe Results			
				Peak Effect			
	ANSYS vs Measured Correlation					Effect of Mass Modeling Problem	
Node	Poor	Fair	Good	1% vs 7% Damping	Typical Supports vs Actual	Typical K	Actual K
166x	✓			100%	90%	20%	53%
166z		✓		260%	27%	31%	17%
77x		✓		90%	67%	10%	100%
77y		✓		167%	38%	20%	34%
77z	✓			210%	64%	0%	6%
36x		✓		250%	17%	10%	10%
36y			✓	200%	88%	0%	0%
36z			✓	240%	26%	0%	5%
3x		✓		220%	65%	12%	12%
3z			✓	220%	51%	12%	4%
58x	✓			300%	0%	20%	
58z	✓			350%	0%	54%	54%
53x	✓			300%	10%	31%	12%
53y		✓		400%	0%	77%	77%
53z		✓		400%	0%	57%	57%

INEL-S-28 057

HDR Comparison Summary

Node	Mass Correction Results			Remarks
	Improved	No Change	Worse	
166x	✓			NUPIPE (7%) compares better than ANSYS
166z		✓		ANSYS compares better
77x	✓			NUPIPE compares better
77y	✓			Corrected NUPIPE has excellent agreement
77z	✓			
36x		✓		
36y		✓		NUPIPE w/corrections & 7% excellent
36z		✓		
3x		✓		ANSYS shifted freq from measured
3z	✓			Nupipe correlation better than ANSYS
58x	✓			
58z	✓			NUPIPE 1% gives good correlation
53x	✓			
53y			✓	NUPIPE Excellent correlation
53z			✓	

INEL-S-28 054

Conclusions

- Current analysis methods predict conservative results
- Best results produced with
  - (a) NUPIPE
  - (b) 7% damping
  - (c) Mass corrections
  - (d) Actual stiffness
- Differences between NUPIPE and ANSYS results
  - (a) Damping scheme
  - (b) Program solution technique
- Analysis methods give reasonable predictions

INEL-S-28 295

NRC FORM 335 (7 77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)
NUREG/CP-0023, Vol. 4

4. TITLE AND SUBTITLE (Add Volume No., if appropriate)
Eighth Water Reactor Safety Research Information Meeting

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)

5. DATE REPORT COMPLETED
MONTH | YEAR

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

DATE REPORT ISSUED
MONTH: March | YEAR: 1982

6. (Leave blank)

8. (Leave blank)

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Same as 9, above.

10. PROJECT/TASK/WORK UNIT NO.

11. CONTRACT NO.

13. TYPE OF REPORT

PERIOD COVERED (Inclusive dates)

15. SUPPLEMENTARY NOTES

14. (Leave blank)

16. ABSTRACT (200 words or less)
This is a compilation of papers which were presented at the Eighth Water Reactor Safety Research Information meeting. It consists of four volumes.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS (This report)
Unclassified

20. SECURITY CLASS (This page)
Unclassified

21. NO. OF PAGES

22. PRICE
$

NRC FORM 335 (7-77)